THE TECHNOLOGY FOR AN ULTRASAFE REACTOR

HEARING

BEFORE THE

SUBCOMMITTEE ON
ENERGY RESEARCH AND PRODUCTION

OF THE

COMMITTEE ON
SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES

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THE TECHNOLOGY FOR AN ULTRASAFE REACTOR

WEDNESDAY, SEPTEMBER 17, 1986

HOUSE OF REPRESENTATIVES,
COMMITTEE ON SCIENCE AND TECHNOLOGY,
SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION,
Washington, DC.

The subcommittee met, pursuant to call, at 9:30 a.m., in room 2359, Rayburn House Office Building, Hon. Marilyn Lloyd (chairman of the subcommittee) presiding.

Mrs. LLOYD. The Subcommittee on Energy Research and Production will come to order.

Good morning to all of you here.

We must recognize that as irrational as it may sound, the Chernobyl accident has had a negative effect on the public perception of the U.S. Nuclear Power Program. Although the Russian accident occurred in a vehicle that was not designed with the safety features of our light water reactors, it has, nevertheless, made the United States public more wary of nuclear technology.

Thus, I believe that it is now even more critical for this committee to push for aggressive development of the passively safe modular high-temperature gas reactor. We have the responsibility to look beyond today's marketplace in which reactor cancellations—not new orders—have been the rule of the day.

The subcommittee has had an increasing dialog with the Nuclear Regulatory Commission regarding the licensing of advanced reactors, and we have published a subcommittee report on that topic. We shall continue to address the regulatory aspects of the MHTGR.

Unfortunately, despite our good faith efforts to reach a consensus in the nuclear community, we are dismayed to hear the DOE talking about facing off the MHTGR against innovative liquid metal reactor [LMR] designs. I must presume that they are forced to this approach as a result of budgetary pressure. However, I would remind the DOE that wishing won't make it so because in programmatic terms such a faceoff makes no sense.

I am alarmed at the lack of consensus regarding the Department's Advanced Reactors Program, and further troubled by the new Assistant Secretary's remarks regarding the advanced light water reactor. If the advanced light water reactor is the chief machine for our nuclear future, then there is little reason to focus on either the MHTGR or the LMR Programs.

Further, since there is no need for Federal funding of advanced light water reactor technology, there is virtually no case for a strong nuclear program.

It appears that doing what seems right to us for the country in nuclear R&D will be extremely difficult. Yet, I firmly believe that this committee has been on the right path for the past 3 years in this post Clinch River era.

I am looking forward today to some good news about the technology advancements in this MHTGR Program, and I expect to get a much better idea what resources will be required to license and demonstrate this design.

Without this focus in the Civilian Reactor Program to match the space nuclear focus chiefly driven by defense needs, we will not have a nuclear program that is defensible at home or abroad.

I appreciate the constructive participation of the private sector and national laboratory people who have decided that this concept is, indeed, worth taking some risks for.

Before we hear from our witnesses, I do want to express my appreciation for the ranking Republican on this committee, Mr. Morrison.

We would like to hear from you at this time.

Mr. MORRISON. Thank you, Madam Chairman.

I just want to comment very candidly that, yes, the Civilian Nuclear Reactor Program is under great pressure. And I think this subcommittee can only follow the path that you are providing, Madam Chairman, and that is to look very, very carefully, very critically at the various outcomes of the research that we are funding, and the directions that we are being led by, the Department of Energy, in these efforts. Because whatever we do decide is going to have to be based on the absolute facts and the programs that have the greatest potential for serving the future need, which I think we all recognize are out there, even though Chernobyl and waste issues, and a host of other items do complicate the picture somewhat.

I think that presents that much more of a challenge to the committee and to the Department to find a proper path and pursue it enthusiastically for the sake of that day when all of a sudden we are ready for a new generation of reactors. I know the look at, particularly HTGR, which has been a favorite of the subcommittee, will be important today.

I hope that we can in the future take a look at some of the other technologies which are also coming along, I think, at a very acceptable pace.

Thank you.

Mrs. LLOYD. Your point is well taken. We do want to take a look at the other technologies as well before this year comes to a close.

Our first witness today is Dr. Wilcox.

We are so happy to have you today. You are the Deputy Assistant Secretary for Reactor Systems Development and Technology at the Department of Energy.

I do understand that this will be your last appearance before this subcommittee. We look forward to hearing from you, and we certainly wish you the very best in the private sector.

Your complete testimony will be made a part of the record. You may proceed as you wish, Dr. Wilcox.

STATEMENT OF LYLE C. WILCOX, DEPUTY ASSISTANT SECRETARY FOR REACTOR SYSTEMS DEVELOPMENT AND TECHNOLOGY, DEPARTMENT OF ENERGY; JERRY D. GRIFFITH, ASSOCIATE DEPUTY ASSISTANT SECRETARY FOR REACTOR SYSTEMS DEVELOPMENT AND TECHNOLOGY, DEPARTMENT OF ENERGY; AND FRANCIS X. GAVIGAN, DIRECTOR, OFFICE OF ADVANCED REACTOR PROGRAMS, DEPARTMENT OF ENERGY

Dr. WILCOX. Thank you very much, Madam Chairwoman. I have enjoyed working with you and being in front of this committee.

Mrs. LLOYD. We hope you will come back to see us.

Dr. WILCOX. I shall do so.

Mrs. LLOYD. Would you identify the gentlemen that you have with you.

Dr. WILCOX. Thank you. The gentlemen that I have with me today, on my right is Mr. Jerry Griffith, the Associate Deputy Assistant Secretary for our Reactor Systems Development and Technology. He has a very strong, long background in nuclear energy, particularly in the safety aspects, and has worked a good bit in the Gas Reactor Program for DOE.

On my left is Mr. Francis (Frank) Gavigan, Director of the Advanced Reactor Programs. Frank directs the activities of all of the advanced technologies for us—gas reactors as well liquid metal, and is deeply involved in the interactions with the NRC and the licensing activities in our program relative to the HTGR.

Mrs. LLOYD. Mr. Griffith, Mr. Gavigan, let me welcome you also.

Dr. WILCOX. Madam Chairman, I have prepared a written testimony. If you permit I would like to go over it for you, read excerpts from it as an introduction to this subject.

Mrs. LLOYD. Without objection.

Dr. WILCOX. Madam Chairman, first, thank you for the opportunity to appear before you today where we will describe the Department of Energy's High-Temperature Gas-Cooled Reactor Program. I shall comment on the future potential of this reactor system and the issues facing its introduction into the domestic energy marketplace. Right at the outset I would like to say that you and your committee have been very effective and continuing supporters of this advanced technology. Your advice, your encouragement, and your support are very much appreciated.

The ultimate objective of the Department's Advanced Reactor Program, in general, is to develop safe, economic, licensable, and reliable reactor systems. To this end the Department is assessing various advanced reactor systems. The HTGR concept is one approach to achieving this objective.

Within the HTGR Program, our goal is to demonstrate that an efficient HTGR can be a viable commercial investment for the production of electric power, high-temperature process heat, or both in a cogeneration mode.

There are certain inherent safety features of the HTGR that can lead to economic benefits by limiting financial exposure and risk of reactor damage and health effects. The modular HTGR Program is utilizing these inherent safety features to develop a system that can protect the health and safety of the public without operator actions, without operation of any safety systems, and without the need to evacuate or shelter anyone beyond the plant boundaries. The design further focuses on providing long operator-reaction time and on providing long operator-reaction time and tolerance for inappropriate operator actions, thus minimizing consequences due to operator-induced malfunctions.

The HTGR incorporates passive safety features that are based on the physical laws of nature, characteristics of materials, and careful disciplined development of plant design and construction features. Four such examples are: One, designing in a large negative temperature coefficient that terminates the nuclear reaction when fuel heats up; two, using fuel and core materials that provide large margins to fission products release; three, emphasizing a low-power density core with a massive graphite structure that normally operates at low temperature and limits rate of heatup; lastly, four, using an innovate core geometry to enhance removal of afterheat by passive conduction and radiation effects.

In the past, traditional economic scaling laws have favored larger plants; however, HTGR evaluations indicate that the small modular plant approach with passive safety features may well provide favorable economics by: Elimination of unnecessary engineered safety systems; nonsafety classification of unnecessary engineered safety systems; nonsafety classification of balance of plant; prelicensed, standardized design; shortened construction schedules; serialized factory fabrication; and multiple modules on a single site. With this approach, the Department's research and development program is attempting to enable low generating costs resulting from lowered initial investment costs and small financial exposure over fewer years. This means lowered overall financial risk for utilities and better long-term rates for the ratepayers.

In 1984, the program was realigned to evaluate the technical and economic feasibility of smaller, passively safe HTGR concepts. The realignment was influenced by this subcommittee's interest in assessing the potential for a small modular system. The realignment was also based on requirements identified by utility users.

The efforts to date have been coordinated with a group of potential utility users, thus assuring that the design and technology programs are structured to meet operating, licensing, and investment protection requirements which they have identified. These utilities also have identified technical requirements which they would like to achieve in a future nuclear plant to make it more attractive to the utility marketplace. Such requirements include: An availability factor of 80 percent; outage factors of less than 10 percent; very low probability 10 to minus fifth, for plant year, of financial investment loss due to plant malfunctions; forced outage rates of less than 10 percent; and a stipulation that no public evacuation or sheltering will be necessary in the event of a reactor plant incident.

These requirements have been incorporated into the design objectives of modular HTGR plants. The reference modular concept that

you will be hearing about, selected in September 1985, is a 350-megawatt thermal plant containing an annular core of prismatic fuel within a steel vessel. Design development, the supporting technological base, and the licensing infrastructure to support deployment are proceeding, taking into account the information obtained from the Soviet incident at Chernobyl.

The program is continuing to focus on designing this specific reference plant while supporting the submission of a preliminary safety information document to the Nuclear Regulatory Commission on September 30, 1986.

The licensing plan for the modular HTGR has been approved by the NRC, and safety criteria specific to the modular HTGR design are being developed. We hope to obtain a licensability statement from NRC early in fiscal year 1988. Licensability is one of the necessary conditions for support of further development of any of the advanced reactor concepts that DOE is pursuing.

Technology development plans for the modular HTGR have been developed and implemented in key areas such as fuel performance, graphite development, and fission product behavior. Work has been initiated to perform tests verifying fuel performance and fission product behavior to support the licensing basis for the modular HTGR.

The Department has continued international cooperation with the Federal Republic of Germany in areas of fuels, alloys, graphite, and very high temperature materials. The Department and the FRG are developing agreements to expand this U.S./FRG program to include cooperative HTGR safety efforts. Also, the Department has entered into a cooperative agreement with Japan for HTGR technology development in the area of fuel performance and has initiated efforts to expand this agreement to include graphite and metal testing programs.

Certain key technical issues must be resolved for the commercialization of modular HTGR's. They are: No. 1, the performance of HTGR fuel to meet stringent offsite dose requirements.

No. 2, the performance of HTGR's air natural circulation reactor cavity cooling system to passively remove decay heat.

No. 3, the performance of advanced helium circulator with magnetic bearings.

No. 4, realistic design criteria for graphite components to remove overly conservative criteria.

No. 5, performance of selected key systems, subsystems, and components.

No. 6, methods development and code verification issues dealing with thermal hydraulics analysis.

No. 7, metals properties areas in terms of the vessel and internal structures.

In conclusion, it is clear that additional work must be completed by the Department and industry before the full potential of the reactor system can be realized. These efforts would also be necessary to define the full funding requirement.

Presently we have a well-organized, disciplined, and cost-effective program, with a proper mix of participants and with appropriate support and experimentation to resolve the remaining uncertainties on an issue-by-issue basis as quickly as is practical. These ef-

forts must provide the foundation for judging the need and defining the schedule for a comprehensive demonstration effort.

Madam Chairman, this completes my statement. We stand ready to answer questions that you may have.

[The prepared statement of Dr. Wilcox follows:]

STATEMENT OF
DR. LYLE C. WILCOX
DEPUTY ASSISTANT SECRETARY FOR REACTOR
SYSTEMS, DEVELOPMENT, AND TECHNOLOGY
U.S. DEPARTMENT OF ENERGY
BEFORE THE
COMMITTEE ON SCIENCE AND TECHNOLOGY
SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION
U.S. HOUSE OF REPRESENTATIVES

SEPTEMBER 17, 1986

Madam Chairman, thank you for the opportunity to appear before you today to describe the Department of Energy's High Temperature Gas Cooled Reactor (HTGR) Program. I shall comment on the future potential of this reactor system and the issues facing its introduction into the domestic energy market place. You and your committee have been continuing supporters of this advanced technology. Your advice, encouragement, and support is appreciated.

The ultimate objective of the Department's Advanced Reactor Program is to develop safe, economic, licensable and reliable reactor systems. To this end, the Department is assessing various advanced reactor systems. The HTGR concept is one approach to achieve this objective. Within the HTGR program, our goal is to demonstrate that an efficient HTGR can be a viable commercial investment for production of electric power, high temperature process heat, or both in a cogeneration mode.

There are certain inherent safety features of the HTGR that can lead to economic benefits by limiting financial exposure and risk of reactor damage and health effects. The modular HTGR program is utilizing these inherent

safety features to develop a system that could protect the health and safety of the public without operator actions, without operation of any powered safety systems and without the need to evacuate or shelter anyone beyond the plant boundaries. The design further focuses on providing long operator reaction time and tolerance for inappppropriate operator actions, thus minimizing consequences due to operator-induced malfunctions.

The HTGR incorporates passive safety features that are based on the physical laws of nature, characteristics of materials, and careful, disciplined development of plant design and construction features. Four such examples are: 1) designing in a large negative temperature coefficient that terminates the nuclear reaction when fuel heats up; 2) using fuel and core materials that provide large margins to fission product release; 3) emphasizing a low power density core with a massive graphite structure that normally operates at low tempertature and limits rate of heatup; and 4) includes an innovative core geometry enhancing removal of afterheat by passive conduction and radiation.

In the past, traditional economic scaling laws have favored larger plants, however HTGR evaluations indicate that the small modular plant approach with passive safety features may well provide favorable economics by: elimination of unnecessary engineered safety systems; nonsafety classification of balance of plant; pre-licensed, standardized design; shortened construction schedules; serialized factory fabrication; and multiple modules on a single site. With this approach, the Department's research and development program is attempting to enable low generating costs resulting from lowered initial investment costs and smaller financial

exposure over fewer years. This means lowered overall financial risk for utilities and better long-term rates for the ratepayers.

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The efforts to date have been coordinated with a group of potential utility users, thus assuring that the the design and technology programs are structured to meet operating, licensing and investment protection requirements which they have identified. These utilities also have identified technical requirements which they would like achieved in a future nuclear plant to make it attractive to the utility market. Such requirements include:

- an equivalent availability factor of at least 80 percent
- a plant outage factor of less than 10 percent
- a very low probability (1×10^{-5} per plant year) of financial investment loss due to plant malfunctions.
- a forced outage rate of less than 10 percent; and

- a stipulation that no public evacuation or sheltering will be necessary in the event of a reactor plant incident.

These requirements have been incorporated into the design objectives of Modular HTGR plants. The reference modular concept selected in September 1985 is a 350 megawatt thermal (MWt) plant containing an annular core of prismatic fuel in a steel vessel. Design development, the supporting technology base, and the licensing infrastructure to support deployment are proceeding, taking into account the information obtained from the Soviet reactor accident at Chernobyl.

The program is continuing to focus on designing this specific reference plant while supporting the submission of a Preliminary Safety Information Document to the Nuclear Regulatory Commission (NRC) by September 30, 1986. The licensing plan for the modular HTGR has been approved by the NRC, and safety criteria specific to the modular HTGR design are being developed. We hope to obtain a licensability statement from NRC early in FY 1988. Licensability is one of the necessary conditions for support of further development of any of the advanced concepts DOE is pursuing.

Technology Development Plans for the modular HTGR have been developed and implemented in key areas such as fuel performance, graphite development and fission product behavior. Work has been initiated to perform tests verifying fuel performance and fission product behavior to support the licensing basis for the modular HTGR.

The Department has continued international cooperation with the Federal Republic of Germany (FRG) in areas of fuels, alloys, graphite and very high temperature materials. The Department and the FRG are developing agreements to expand the U.S./FRG Program to include cooperative HTGR safety efforts. Also, the Department has entered into a cooperative agreement with Japan for HTGR technology development in the area of fuel performance and has initiated efforts to expand this agreement to include graphite and metal testing programs.

Certain key technical issues must be resolved for the commercialization of modular HTGRs. They are:

1. Performance of HTGR fuel to meet stringent offsite dose requirements.

    - o Demonstrate and verify key aspects of HTGR fuel performance under accident conditions including fission product retention and transport characteristics.

    - o Verify that the laboratory scale LEU fabrication process developed to meet more stringent offsite dose requirements is applicable to fabrication of production quantities.

2. Performance of HTGR's air natural circulation reactor cavity cooling system to passively remove decay heat.

    - o Develop and verify code for predicting system performance

- o Demonstrate passive heat removal performance of the system

3. Performance of advanced helium circulator with magnetic bearings (the water bearings in the circulator have been problem components at Fort St. Vrain)

   - o Develop and demonstrate magnetic bearings technology

   - o Perform integrated test demonstrating performance and reliability of circulator, motor and their bearings

4. Realistic design criteria for graphite components to remove overly conservative criteria

   - o Develop new design criteria and demonstrate ability of graphite components to meet criteria.

5. Performance of selected key systems, subsystems and components

   - o Perform seismic tests to verify performance of graphite components

   - o Demonstrate HTGR Fuel Handling to provide basis for planned outage predictions

- o Perform tests on steam generator tube bundle assembly to demonstrate seismic performance and performance of bimetallic welds in HTGR environment

- o Verify hot duct performance predictions

- o Verify that the defense-in-depth provides the required containment functions without the need for a conventional containment building.

6. Methods development and code verification issues

- o Validate thermal hydraulics code to demonstrate ability to predict core flow, by-pass flow, and temperature distributions in the core, and steam generator and shutdown cooling heat exchanger.

- o Develop and demonstrate methods required for in-service inspection of reactor vessel, hot duct, steam generator and reactor cavity cooling system.

- o Develop simulation code and demonstrate acceptable performance of automated control system and plant protection system for multiple HTGR modules

7. Metals properties issues

- o Provide materials data to verify acceptable performance of vessel and internal structures under normal and accident conditions and extend ASME code to include higher temperature operation.

In conclusion, it is clear that additional work must be completed by the Department and industry before the full potential of this reactor system could be realized. These efforts would also be necessary to define the full funding requirement. We have a well organized, disciplined and cost effective program. There is a proper mix of participants, with appropriate support and experimentation to resolve the remaining uncertainties on an issue-by-issue basis as quickly as practicable. These efforts must provide the foundation for judging the need and defining the schedule for a comprehensive demonstration effort.

Madam Chairman, this completes my statement. My colleagues and I would be pleased to respond to your questions at this time.

Mrs. LLOYD. Thank you very much, Dr. Wilcox.

In view of the funding level that we have been able to obtain for this program, because this committee certainly believes in the MHTGR Program; do you think that it is really progressing at the level it should with the annual funding that has been provided?

Why have—to be more specific—why haven't we completed the conceptual design at this point?

Dr. WILCOX. In answer to your first question, and in terms of the funds that have been provided and the work that has been done, I think a great deal has been accomplished. Later today you will hear a great deal in terms of specifics of what has been done.

I think it is terribly important to realize that what we are talking about here is a brandnew kind of system. It is not simply a small step from a present system. But it is a new idea, a new concept, a new size, a new methodology for manufacturing. These and all of the other factors that you will hear about today, mean that this is going to take time.

I think that the money that has been spent to date has moved an original idea from that point to a point now where we see specific issues developing, we see specific technologies developing, we see solutions to problems, and we see some problems that remain to be solved.

The list of progress through 1985 and 1986 can be provided to you, but I think it is impressive in terms of the work that is completed.

Mrs. LLOYD. Have the engineering drawings been completed, Dr. Wilcox?

Dr. WILCOX. A number of engineering drawings and data have been completed. A few pieces of data that I have for you, conceptual designs, for example, realizing that this started just in September 1985, and is nearly 60 percent complete now. That is good progress, I feel.

In terms of design reports, there are over three dozen of those design reports that have been produced, distributed. There are over 110 engineering drawings with details, considerable details.

Mrs. LLOYD. Has the core design been completed?

Dr. WILCOX. Core design has been completed, including the specifications for fumes, fueling loading enrichment and the distribution of the fuels.

Mrs. LLOYD. What enrichment level is this design based on?

Dr. WILCOX. 19.8 percent.

Mrs. LLOYD. How much of the steam supply system has been completed?

Dr. WILCOX. Approximately 60 percent of that.

Mrs. LLOYD. What heat transfer computer programs have been used to model the natural cooling efficiency?

Dr. WILCOX. There have been a number of them used, probably somewhere around 200 different codes have been used or have been developed for use on heat transfer and thermal characteristics, hydraulic characteristics for HTGR.

To indicate, there are about seven codes which are directly related to heat transfer processes in various parts of this plant. That is a lot of computer simulation.

Mrs. LLOYD. Dr. Wilcox, one of the things that concerns me as chairman of this committee is that I don't want us to see a lack of congressional support if we don't make the necessary progress with our HTGR Program. Certainly, you have my support and Mr. Morrison's support. But we certainly need a larger backing by the Congress. And I see this slipping if we don't move ahead with this program.

Is the DOE really committed to this reactor, to the design, the research, and the testing, more than just paper?

Dr. WILCOX. Yes; I think DOE is first committed to the concepts that we are talking about in advanced reactor design. We certainly are committed at the level of looking at the conceptual designs, looking at those features that are going to have to be proved, those technologies that will have to be supporting these designs. And we are committed to looking carefully at the relationship, the advantages, and disadvantages of the different types of designs, whether they are liquid metal, or gas, or advanced light water designs.

Mrs. LLOYD. What is the extent of the work that has been contracted out by the Department to the national laboratories or other contractors?

Dr. WILCOX. A great deal of the work done here, virtually all of it, as a matter of fact, of course, is contracted out. The national laboratory work—Jerry, do you have some of those figures—about $3 million to, primarily, Oak Ridge.

Mrs. LLOYD. Do you feel the work is progressing on schedule, or have there been schedule slippages?

Dr. WILCOX. The work is on schedule. The planning that has been done 1 year ago, 2 years ago, much of that work, except for a few minor issues is on schedule, particularly the licensing interactions.

Mrs. LLOYD. Can you give us an accounting of what funding is really needed by year for us to complete the design the testing and the licensing of the reactor?

Dr. WILCOX. I need to qualify some of the figures, because as we go through this project it is part research, part development, part engineering. As you move through it you find certain parts of the program that you want to speed up, and certain parts of the program where you have to do additional testing.

So it is a little difficult to predict out in 8 or 10 years and say exactly it is going to cost by year, and what areas are going to have to receive the largest amount of support.

Mrs. LLOYD. This committee is committed to a close oversight, to not allow for slippages in this schedule. I think it is very important that this committee is provided more information as to the schedule, and where we are, and what you think are realistic expectations, and realistic requirements for the program as well.

What are the remaining tasks to be completed before this program can move to licensing completion?

Dr. WILCOX. Francis, would you like to summarize those for us?

Mr. GAVIGAN. Before we get finished with NRC and their interaction, we will have conducted a number of meetings through fiscal year 1987. In particular, three meetings of major importance are being scheduled by NRC as a result of information meetings that have been conducted with the LMR and HTGR people.

Three issues in particular, which I think will be addressed later, one having to do with standardization and certification of a prototype. The second with a source magnitude of the size and magnitude of the source term. The third is the issue of the containment/confinement systems.

These are being addressed now by the regulatory people and will be brought up with the commissioners as a matter of great interest that affects all of the reactors. Given that as a successful mode of operation, we anticipate receiving a licensability letter from NRC at the end of 1988—1987, excuse me.

Dr. WILCOX. Or the first quarter of 1988.

Mr. GAVIGAN. Right, or the first quarter of 1988.

Mrs. LLOYD. Thank you very much.

Mr. Morrison.

Mr. MORRISON. Thank you.

Dr. Wilcox, we are going to miss you. Good luck from all of us that have had the opportunity to work with you through these past few years.

How much international cooperation—you have mentioned a couple of things in your basic testimony—could you expand on that; what is the cooperative investment, for instance, that is being made in this total effort?

Dr. WILCOX. I would like to preface that remark by saying that just last year, on our visit to Japan, we negotiated the umbrella agreement—with JAERI—the equivalent parties in Japan to DOE that are doing gas reactor development work, and one of the comments that was made after the signing of that agreement is that at long last after 10 years we have completed this vehicle though which we can do cooperative work in gas reactor technology.

I think that was a milestone. And we will try to build on that kind of a relationship.

The cooperative programs that are ongoing now, as I mentioned that with Germany for a fuel development and graphite development, were fission product transfer, and transport technologies, metals, metal characteristics and properties, and reactor safety studies in the association with Japan. This deals primarily with fuel irradiation studies, graphite technology, and additional work in core instrumentation, which is obviously very important in this kind of reactor.

In the exchange with Germany, since its start in 1978, we value that at something in the order of 10 million dollars' worth of work that has been done cooperatively, that we have gained from that association.

Further development in the testing of fuels from 1981 through approximately 1990, we feel that there is about $60 million worth of cooperative work that would be done in that arena.

With Japan, this is a more recent exchange. This was done in 1986, as I indicated, and involves in 1986, about $0.5 million, perhaps, closer to $400,000. We project that over the next 5 years about $2 million of work will be done cooperatively with Japan.

Mr. MORRISON. How does this compare with research, cooperative research into other technologies?

Dr. WILCOX. The type of liquid metal, or——

Mr. MORRISON. Yes; most other work that is going on?

Dr. WILCOX. In technologies and light water area, that is mainly individual private sector to private sector work. In the case of liquid metal and breeder, this would be less than would be ongoing with liquid metal or breeder reactor development, designs and technologies, primarily because that has a long history of an association between Japan and the United States through——

Mr. MORRISON. Do you get any feeling from these international efforts that they have made choices, or are they sort of like we are and that is pursuing several technologies, and knowing the time, tide, and circumstances will lead to some conclusions, ultimately?

Dr. WILCOX. Certainly the technologies that are being looked at by most countries, in terms of a significant investment, would be liquid metal and gas reactor technologies. I think they are comparing these two in terms of their purpose and function.

In terms of what will finally be adopted, I think each country has its own set of conditions that is going to determine what kind of a reactor, the size, the public perception, and so on, that will help determine all of that.

Mr. MORRISON. I get the impression, and maybe it is wrong—I haven't been here that long—but I get the feeling that some of the enthusiasm for HTGR is driven by the budget process, that is there is sometimes a lack of enthusiasm through OMB, perhaps, even reflecting some attitudes within the Department of Energy, and that enthusiasm gets reinstated when it gets up here on the Hill; is that observation inaccurate?

Dr. WILCOX. I will clarify it this way, the enthusiasm is always, I guess, the pressures that are applied, one of which is budget, upper bounds on budget. Certainly there have been considerable pressures to make sure that we didn't—that is DOE—go off in a whole host of different directions with big reactors, little reactors, different technologies. And I think we have attempted to consolidate that as much as possible in the last 2 years. But I would say that everyone has their technology choice, their favorite from an engineering or a scientific point of view, or political, and I guess that is to be expected.

But I think the fact that the DOE program looks to running advance reactor research that has this inherent safety characteristics, regardless of reactor, is something that is important and shouldn't be overlooked. This draws—these different designs, and different technologies to a common base that I think is helpful and good.

Mr. MORRISON. With the path that we are currently on with these competing technologies—and there are several, in which obviously the committee has an interest—are we marching in a direction and with a timetable that will provide the results that we are going to need in this country as far as our energy futures are concerned; does your work coincide, on the pace that you are moving with what we might project the need for the new orders we hope come in within the next decade?

Dr. WILCOX. I think it does. And it provides alternatives. What we are attempting to do, of course, is move into a state of readiness, so that when this need does occur we are not going all the way back to reassessing all kinds of technologies and all kinds of options.

I think that in providing that readiness you look at different technologies and different options. For example, in the gas reactor option, you have a characteristic of being able to operate the system at very high temperatures.

In the case of liquid metal, you have the option of breeding, which the gas reactor does not efficiently provide for you. And keeping these alternatives ready and providing the research to carry them out, I think is a good plan in strategy. And I think in terms of uncertainty of the date at which the requirement for this type of device is going to be needed, when the commercial sector is going to have to provide this electricity, that is as good a plan as I can think of.

Mr. MORRISON. Thank you. I hope you are right.

I think the efforts of this subcommittee will focus along those same lines as well.

Again, good luck to you, sir.

Thank you, Mrs. Chairman.

Mrs. LLOYD. Thank you, Mr. Morrison.

Mr. Fawell.

Mr. FAWELL. Thank you, Madam Chairman.

In light of Chernobyl, I certainly applaud the continued R&D in regard to inherently safe advance reactors. I am relatively new, so many of my questions may betray my ignorance, but insofar as the modular HTGR is concerned, you had indicated that it really is sort of a rebirth in concept and design, and you made reference to the new size, it is a smaller reactor; that is correct?

Dr. WILCOX. That is right.

Mr. FAWELL. Why is it a smaller design?

Dr. WILCOX. A clarification, in terms of what sometimes is confusing, there is modular versus smaller design of a reactor system. We are talking modular, which means that several modules can be put together to create a large system, if you wish to do that.

The advantages that we have seen in reducing the size of the modules come mainly from how these are built, and we feel that if they are small enough they can be factory fabricated and shipped to the sites, eliminating a fair amount of field work, which is delicate work in reactor systems. We also feel that this would be a type of configuration that would allow a utility company to move upward in the scale progressively, as opposed to taking one big bite, 2,000 megawatts or something of that type. And I believe scientifically and from an engineering and technical point of view, characteristics have been revealed in the design of these small systems that allow for the inherent safety of all that to be achieved in a little different way, a little more effectively than in the larger systems.

So all of these factors have encouraged us to keep on this particular engineering track.

Mr. FAWELL. Is the modular aspect to which you have made reference, is that also a possibility in other technologies which have inherent safety as a factor?

Dr. WILCOX. In terms of nuclear systems?

Mr. FAWELL. Yes, liquid metal, for instance, is the modular——

Dr. WILCOX. Yes, the liquid metal and gas reactor designs that we are looking at are both small modularized configurations; that is right.

Mr. FAWELL. What about light water reactors?

Dr. WILCOX. There is also work being done, a study being run on the small size, light water reactors, and the implications of designing those.

Mr. FAWELL. There is a technology called, IES, I guess that is the acronym; is DOE doing any work in that regard?

Dr. WILCOX. Not to my knowledge at this point, I think not.

Mr. FAWELL. If it is modular insofar as the HTGR is concerned, and say, liquid metal reactors, can there be large reactors also in the same inherently safe features be retained?

Dr. WILCOX. Well, our feeling in looking at the smaller configurations is this is where we are seeing the benefits in the inherent safety. But I cannot state at this point that you could not build a large reactor with inherently safe characteristics as well.

But we simply see the efficiencies and the economics and the other advantages around the smaller modularized systems to be so beneficial that we are going to concentrate our efforts there.

Mr. FAWELL. As I look at the new technologies, and look at safety, and cost efficiency, and then also the waste products—in other words, the fuel reprocessing design that can work coordinately with the inherent safety features; how does that fit insofar as the HTGR is concerned; can we fuel reprocessing so that we will have less volume of hazardous waste, and less—is called higher isotopes—of your waste product, perhaps, eliminating to the degree possible plutonium grade weapons potential in the form of the waste product?

Dr. WILCOX. You raise an extremely important and relevant issue, and that is as we look at all of these reactors designs, right along with the engineering design of the reactor, steam generators, the system in general, and the economics, we clearly are going to have to pay considerable attention to the waste stream, the handling of the radioactive materials, the spent fuels.

In the case of liquid metal systems, we certainly are doing that with some of the ANL work that is currently a part of the liquid metal reactor systems.

Long-life cores, and fuels are being studied. That is cases where you would fuel a reactor and would not expect to fuel again for several years, 3 years, 5 years, 8 years.

Then, of course, reprocessing, what the reprocessing cycle will look like, what the waste stream will look like, and how much it will cost.

In the case of the gas reactors that you specifically asked for, we are not looking at reprocessing technology, at this point, in the kind of detail that we are in the liquid metal systems. It is a once-through fuel cycle.

Mr. FAWELL. That would seem to me to be something that, perhaps, one ought to look at. The safety features and the costs are tremendously important in terms of proliferation of waste. As I sat in other meetings on the two subcommittees of which I am a member, and I look at the technology of getting rid of the hazardous waste, which, basically, is digging a big hole in the ground and

drop it there. It doesn't impress me a great deal, as a layman, now maybe it is good. But there ought to be, as we talk about recycling of waste for the average American home, which we ought to be certainly looking at, it seems to be that as a part of all of our energy sciences we ought to be vitally concerned about recycling any waste and utilizing that, what I would assume to be portions of the matter which comes out as waste product, and using them; rather than dumping them some place where whole States and regions become understandably concerned—and all the political fallout.

So I would hope that as we work on the various advanced nuclear reactors that is a prime ingredient that we continue to look at.

Dr. WILCOX. I assure you that is the case.

Mr. FAWELL. I don't know what PIUS is doing, that I think was invented by the Swedes, as I recall——

Dr. WILCOX. Sweden.

Mr. FAWELL. But not only the feature of inherent safety which Chernobyl has taught us, we better be concerned about even if we never build another advanced nuclear reactor in this country. Because two-thirds of the advanced nuclear reactors, I gather, are out there in the prevailing wind system, which if they build as they have been building in the Soviet Union, God help us.

We ought to be looking at, certainly, the inherent safety feature, and certainly cost. If it isn't cost efficient practically and politically it is not going to be possible.

But then also this latter feature to which I made reference, being able to not pollute in terms of the ultimate waste product.

If we had that, we certainly would have, I think, made a tremendous contribution to the rest of the world, not to ourselves, because maybe we are not going to be building any more nuclear plants.

But to your knowledge PIUS the design, there is no——

Dr. WILCOX. Reprocessing work being done there.

Mrs. LLOYD. Will the gentleman yield?

Mr. FAWELL. Yes.

Mrs. LLOYD. Dr. Wilcox, is it not correct, the PIUS reactor has never been built, it is strictly a paper study at this time?

Dr. WILCOX. Yes, that is a conceptual paper study.

Mr. FAWELL. Thank you, that is all.

Mrs. LLOYD. Thank you very much.

Mr. Stallings.

Mr. STALLINGS. Thank you.

Dr. Wilcox, the summary conceptual design status, September 1986, indicates that the preliminary design will take 4 years to complete; that seems to be excessive, why would it take that long?

Dr. WILCOX. In general, preliminary designs for reactor systems as complicated as this usually run about 3 years, 4 years. Of the 4-year figure that you quote, I think probably 1 year of that would be the final year of the conceptual design, followed by 3 years of preliminary design.

That is a long time, but when you see in subsequent testimony, the complexity of these systems, it becomes quite clear that that kind of time interval is required.

Mr. STALLINGS. When is the prototype reactor scheduled for start of construction?

Dr. WILCOX. Again, I have to preface by saying, if everything goes well, the things that we are planning on prove out to move on schedule, with no hitches, about the mid-1990's, about 1993, perhaps.

Mr. STALLINGS. How can you start long-lead procurement in December 1988, as the schedule indicates, if the preliminary design has not been completed and the license application in the early stages?

Dr. WILCOX. I think that in the first situation, as the design proceeds, there are certain things where you have a very high degree of confidence. First of all, overall in the design, you have a high degree of confidence since you are going to be able to do it, to build it.

Once reaching that point, then there are different stages, different elements of that design, where you know pretty much what you are going to have to do and how you are going to have to do it.

You simply order the long-lead items, and build that part of the system ahead of subsequent parts of the systems where you may be doing structural tests, or fuel tests, or waiting for confirmation data. So it is not a start on a particular date, and then complete the whole design; it is a stage design.

In the case of licensing, you always, I think in R&D, tend to try to move ahead as quickly as you can, if you waited for everything to be absolute, you may never start building a system. So you have to take your best look at the risks associated with the issues left in licensing and then proceed from there; that is what we would have to do.

Mr. STALLINGS. I see.

This committee has been strongly supportive of the advanced reactor R&D programs, which includes both liquid metal and the high temperature gas reactor. As you know, the Idaho National Engineering Laboratory, located in my district, is involved with both of these nuclear technologies; could you comment on how the development objectives, such as the inherent safety of these two systems compare?

Dr. WILCOX. Fundamentally they are the same thing. That is we are looking at systems that have inherent characteristics, thermal and hydraulic characteristics that don't require external powered systems; that don't required layers of external control systems; that have new sensors, smart sensors, and computers and artificial intelligence to help to control these; and that is something of a common umbrella that is used in both of these reactors systems that we are discussing this morning.

In a sense, that makes our program efficient in that we are not moving in diverse directions with big on the one hand and little on the other, or inherently safe in one, and something other than that in the second. I think that is a positive attribute of the focus of the program at this point.

Mr. STALLINGS. Thank you.

Mrs. LLOYD. Thank you very much, Mr. Stallings.

Dr. Wilcox, thank you again for being with us and your associates. We do wish you well.

Dr. WILCOX. Thank you very much, Madam Chairman.

Mrs. LLOYD. Our next witness this morning is Mr. Themis Speis, from the nuclear reactor. As you know we have been very interested in the NRC's policy on our advanced nuclear reactors.

We are looking forward to your comments, Dr. Speis, on the MHTGR licensing process, really what is happening.

You may proceed as you wish. Your entire statement will be made a part of the record.

STATEMENT OF THEMIS P. SPEIS, DIRECTOR OF DIVISION OF SAFETY REVIEW AND OVERSIGHT, OFFICE OF NUCLEAR REACTOR RELATIONS, NUCLEAR REGULATORY COMMISSION, ACCOMPANIED BY THOMAS KING, SECTIONS LEADER, SAFETY PROGRAM EVALUATION BRANCH, NUCLEAR REACTOR RELATIONS, NUCLEAR REGULATORY COMMISSION

Mr. SPEIS. Thank you, Madam Chairman.

On my right I have Mr. Tom King, he is the supervisor of the group that was responsible for the license of both the advanced HTGR and the LMR's.

I would like to read some of my testimony, if you will permit me so?

Mrs. LLOYD. Very good.

If you would speak into the mike, it would be helpful.

Mr. SPEIS. I am pleased to appear before you and your subcommittee today to discuss nuclear reactor activities with respect to the advanced high temperature gas reactor, HTGR. Other members, as I said already, of the NRC staff involved in the advanced HTGR work accompany me today.

I would like to begin my testimony by briefly providing you with some background regarding our activities with the advanced HTGR, and then proceed into our future plans in this area.

Over the past several years there has been renewed interest among the Department of Energy and industry in the development of an advanced HTGR concept. This renewed interest has led to requests for NRC's involvement early in the design process with a goal of developing licensing guidance prior to the submittal of a formal application for a license.

In recognition of this interest, the Commission has developed and issued a policy statement on the regulation of advanced nuclear power plants, budgeted resources and established a group to serve as the NRC focal point for future interactions on advanced reactors.

Over the past 2 years we have been in contact with the Department of Energy and its contractors regarding their efforts to select and develop an advanced HTGR concept, including plans for future interactions with NRC on the advanced HTGR. This has provided us with an opportunity to identify what we consider the key safety and licensing issues associated with this concept.

Interactions have taken place on some of these key safety and licensing issues, on the concept selection, on the proposed top level criteria for the HTGR and on the content and structure of the preliminary safety information document being prepared by the Department of Energy and its contractors for the selected advanced HTGR concept.

Feedback has been provided to the Department of Energy regarding these items discussed above. In addition, two briefings have been held for the Advisory Committee on Reactor Safeguards to familiarize them with the advanced HTGR concept.

I will now say a few things about our future plans. Beginning in early fiscal year 1987 our review will concentrate on the PSID for the advanced HTGR. As the representatives of the Department of Energy just said, they have targeted to provide us these PSID's toward the end of September.

The main purpose of our review would be to establish licensing criteria for the advanced HTGR and to make an assessment of the potential for the proposed design to meet these criteria. The licensing criteria will be established by building up on existing light water reactor, where applicable, and by developing additional criteria, as necessary as to address the unique aspects of the HTGR.

We expect the HTGR designers to propose those criteria which they believe apply to their plant. The Commission is also receptive to proposals for the use of nonprescriptive or performance based criteria which allow for flexibility in the use of the inherently safe features of the HTGR while retaining engineering judgment and defense in-depth to account for uncertainties.

We plan to assess the potential of the advanced HTGR design to meet the licensing criteria by independently evaluating the performance of the key safety attributes of the design. The plant response to a full range of accident conditions will be analyzed.

We expect that this will identify the failure thresholds where accident consequences become unacceptable as well as identify weaknesses in calculational methods and base technology supporting the design. This will then be discussed with the designers and the resolution documented in our safety evaluation report.

Early in the review, we plan to interact with DOE and its contractors on three key policy related topics associated with the advanced HTGR. These three topics are: Treatment of severe accident for the HTGR, including source term and emergency planning; adequacy of containment/confinement plans; and plans for standardization, including extent of plant to be reviewed, depth of standardization, operating experience/R&D required before the Commission would approve or certify as a standard design a new concept.

With respect to standardization, the Commission is presently preparing a policy statement on standardization which would provide additional guidance for pursuing this objective in the HTGR Program.

Other Commission actions which will directly influence our review of the advanced HTGR have also been taken over the past year. These are: issuance of a severe accident policy statement on August 1985; issuance of a final advanced reactor policy statement on July 1986; issuance of a safety goal policy statement on August 1986; and finally, work directed toward future issuance of a revised policy statement on standardization.

These policy statements provide general guidance applicable to all types of power reactors and must be factored into our review of the advanced HTGR.

In summary, it has been and continues to be the Commission policy to pursue activities consistent with programs adopted by the

executive branch, Congress, or focused private sector efforts. Accordingly, NRC has underway activities in preparation for conducting a review of an advanced HTGR conceptual design consistent with DOE industry initiatives.

This activity is intended to provide early guidance to the HTGR designers on safety and licensing issues and to lead to the technical base for future NRC review and research efforts associated with the advanced HTGR. For the subcommittee's information, I have appended to my statement a list of key safety and licensing issues of the modular HTGR.

Madam Chairman, this completes my prepared statement. We would be pleased to answer any questions you may have.

Thank you.

[The prepared statement of Mr. Speis follows:]

PREPARED TESTIMONY

SUBMITTED BY

UNITED STATES NUCLEAR REGULATORY COMMISSION

PRESENTED BY

DR. THEMIS P. SPEIS, DIRECTOR

DIVISION OF SAFETY REVIEW AND OVERSIGHT

OFFICE OF NUCLEAR REACTOR REGULATION

TO THE

SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION

COMMITTEE ON SCIENCE AND TECHNOLOGY

UNITED STATES HOUSE OF REPRESENTATIVES

CONCERNING NRC's ACTIVITIES ON THE ADVANCED HTGR

SUBMITTED: SEPTEMBER 17, 1986

Thank you Madam Chairman. I am pleased to appear before you and your Subcommittee today to discuss Nuclear Regulatory Commission (NRC) activities with respect to the advanced high temperature gas reactor (HTGR). Other members of the NRC staff involved in the advanced HTGR work accompany me today. I would like to begin my testimony by briefly providing you with some background regarding our activities with the advanced HTGR and then proceed into our future plans in this area.

BACKGROUND

Over the past several years there has been renewed interest among The Department of Energy (DOE) and industry in the development of an advanced HTGR concept. This renewed interest has led to requests for NRC's involvement early in the design process with a goal of developing licensing guidance prior to the submittal of a formal application for a license.

In recognition of this interest, the Commission has developed and issued a Policy Statement on the Regulation of Advanced Nuclear Power Plants, budgeted resources and established a group to serve as the NRC focal point for future interactions on advanced reactors.

Over the past two years we have been in contact with DOE and its contractors regarding their efforts to select and develop an advanced HTGR concept, including plans for future interactions with NRC on the advanced HTGR. This has provided us with an opportunity to identify what we considered the key safety and licensing issues associated with this concept.

Interactions have taken place on some of these key safety and licensing issues, on the concept selection, on the proposed top level criteria for the HTGR and on the content of the Preliminary Safety Information Document (PSID) being prepared by DOE and its contractors for the selected advanced HTGR concept. Feedback has been provided to DOE regarding these items. In addition, two briefings have been held for the Advisory Committee on Reactor Safeguards to familiarize them with the advanced HTGR concept.

FUTURE PLANS

Beginning in early FY 1987 our review will concentrate on the PSID for the advanced HTGR, its supporting Probabilistic Risk Assessment and Base Technology Program. The main purpose of our review will be to establish licensing criteria for the advanced HTGR and to make an assessment of the potential for the proposed design to meet those criteria. The Office of Nuclear Reactor Regulation will take the lead for the review with support from the

Office of Nuclear Regulatory Research. The licensing criteria will be established by building upon existing light water reactor (LWR) criteria, where applicable, and by developing additional criteria, as necessary to address the unique aspects of the HTGR. We expect the HTGR designers to propose those criteria which they believe apply to their plant. This has been the case with the HTGR Top Level Criteria which were proposed by DOE and reviewed by NRC. The Commission is receptive to proposals for the use of non-prescriptive or performance based criteria which allow for flexibility in the use of the inherently safe features of the HTGR while retaining engineering judgment and defense-in-depth to account for uncertainties.

We plan to assess the potential of the advanced HTGR design to meet the licensing criteria by independently evaluating the performance of the key safety attributes of the design. The plant response to a full range of accident conditions will be analyzed. We expect that this will identify the failure thresholds where accident consequences become unacceptable as well as identify weaknesses in calculational methods and base technology supporting the design. These will then be discussed with the designers and the resolution documented in a Safety Evaluation Report. Our review will also give consideration to appropriate domestic and foreign HTGR experience.

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Early in the review, we plan to interact with DOE and its contractors on three key policy related topics associated with the advanced HTGR. These three topics are:

- treatment of severe accidents for the HTGR, including source term and emergency planning,
- adequacy of containment/confinement plans, and
- plans for standardization, including extent of plant to be reviewed, depth of standardization, operating experience/R&D required before the Commission would approve or certify as a standard design a new concept.

With respect to standardization, the Commission is presently preparing a policy statement on standardization which would provide additional guidance for pursuing this objective in the HTGR program.

The NRC staff plans to prepare papers for the Commission by mid-FY 1987 on each of these topics recommending a position on each and soliciting Commission guidance. Additional meetings with the Advisory Committee on Reactor Safeguards will also be held to obtain ACRS' views on the design.

Other Commission actions which will directly influence our review of the advanced HTGR have also taken place over the past year. These are:

- Issuance of a Severe Accident Policy Statement on August 8, 1985.
- Issuance of a final Advanced Reactor Policy Statement on July 8, 1986.
- Issuance of a Safety Goal Policy Statement on August 4, 1986.
- Work directed toward future issuance of a revised Policy Statement on Standardization.

These policy statements provide general guidance applicable to all types of power reactors and must be factored into our review of the advanced HTGR.

SUMMARY

It has been and continues to be Commission policy to pursue activities consistent with programs adopted by the Executive Branch, Congress or focused private sector efforts. Accordingly,

NRC has underway activities in preparation for conducting a review of an advanced HTGR conceptual design consistent with DOE/industry initiatives. This activity is intended to provide early guidance to the HTGR designers on safety and licensing issues and to lead to a technical base for future NRC review and research efforts associated with the advanced HTGR. For the Subcommittee's information, appended to my statement is a list of key safety and licensing issues for the modular HTGR.

Madam Chairman, this completes my prepared statement. We would be pleased to answer any questions you may have.

Key Safety and Licensing Issues for the Modular HTGR

- Treatment of Severe Accidents (including source term and emergency planning).
- Containment/Confinement design
- Multi-module control scheme.
- Design and reliability of shutdown systems.
- Design and reliability of decay heat removal systems.
- Provisions for inservice inspection (including graphite structures).
- Use of non-safety grade balance of plant (including control room).
- Implementation of standardization.

Mrs. LLOYD. Thank you very much, Dr. Speis.

I was really anxious to know if any laboratory tests had been conducted by the NRC to establish the behavior of graphite under varying temperatures and moisture conditions?

What has been done; and what information do you need in this area, Dr. Speis?

Mr. SPEIS. Tom, can you address this?

Mr. KING. Yes, let me address that.

The NRC has had underway, probably for the past 4 or 5 years at Brookhaven National Laboratory, a program funded by the Office of Research, one aspect of which looked at graphite, and actually conducted experiments, injecting moisture and air on hot graphite to determine what the reaction was in terms of its rate, the reaction products, the conditions required to react graphite. That program was terminated in 1984, I believe.

As part of our review of the advanced HTGR, we plan to use that data, and if it turns out that we still need additional data, we will identify what we need and, hopefully, fund that beginning in 1988.

Mrs. LLOYD. Has the Chernobyl accident caused any renewed interest in graphite research?

Mr. SPEIS. I think we will have to wait until we get the PSID, and review the design, and see what type of R&D activities have been pursued and completed by DOE, and then we will see if there is any need for NRC to pursue any additional activities, which really fall into the confirmatory arena; that is part of our responsibilities.

Mrs. LLOYD. But, what is your opinion, Dr. Speis?

Mr. SPEIS. Well, the Chernobyl accident, we have been looking at it very carefully. Our initial estimate is that it is not related that much to our reactors. So, I don't think we have to take any radical steps as far as making any changes go.

Mrs. LLOYD. Do you see no relation to our program, then, in the activities of Chernobyl?

Mr. SPEIS. We will have to take a closer look to make sure that our initial assessment is correct.

Mrs. LLOYD. Very good.

I was wondering how you can use light water reactor criteria for the gas reactor licensing; where do you find this parallel, when we have inherently safe reactor, that doesn't require operator actions, or any powered safety system to produce the safety features that are needed and mandated?

Mr. SPEIS. Well, there are many areas that are really similar; and there are many areas that are unique. For example, quality assurance programs, they are basically the same. Evaluating the earthquake characteristics of a site, they are the same.

The criteria, therefore, are similar regardless whether you site an HTGR or a light water reactor. Of course, you have to make sure that the one that you design on the HGTR, it properly accommodates the earthquake forces that are relevant to that particular site.

Now the unique differences of HTGR's, we are talking about the graphite, we are talking about the high temperatures operating conditions, and in those areas you have to develop special and unique criteria to cover those areas; so it is a mix in both.

Mrs. LLOYD. Do you feel that the NRC is moving fast enough in the time scales to move on with this reactor concept?

Mr. SPEIS. I think so. The answer is yes, the commission has allocated the resources. We have requested our oversight committees and the budget committees to provide the appropriate funds. And we have a group at NRC, which is directed by Tom, to review the HTGR as soon as the information is provided to them.

Mrs. LLOYD. I am sure you can get from my opening remarks that I think that we must not lose sight of commitment to the HTGR Program, that we do want to move as rapidly as we can and not waste time, because we certainly feel that the time is now for the MHTGR, because it does have passive safety features, and we hope that you will share the concern of this committee with your colleagues that we do not waste time, that there is no foot dragging on this.

Thank you very much for being with us.

Mr. SPEIS. Thank you, Madam Chairman.

Mrs. LLOYD. Our next witness is Mr. Linden Blue, who certainly is a good friend of this committee. Linden Blue with his brother Neal is the coowner of GA Technologies Corps.; he is certainly a strong supporter of the HTGR Program.

It certainly good to see you. We are interested in your perspective from a successful businessman perspective, and the prospects that you see ahead for the MHTGR Program.

Please proceed.

STATEMENT OF LINDEN BLUE, VICE CHAIRMAN OF THE BOARD, GENERAL ATOMIC TECHNOLOGIES, SAN DIEGO, CA

Mr. BLUE. Thank you, Madam Chairman.

I have to admit that some people who know about the health of the nuclear power business, when they heard we were buying GA, they said, why would you want to do that sort of thing. We really have some very fundamental answers to that.

First of all, GA is quite a diverse company. You know about the fission and fusion businesses. We also have chemical demill, toxic waste disposal, electricmagnetic imagery, electronic sensors, a number of diverse technologies that give GA quite an unusual capability.

But I must say that the one that is most exciting is the modular HTGR. That isn't because we are physicists; we are not physicists. We are businessmen with a bent toward international economics perhaps.

But what we have done is looked at the macroworld picture for energy requirements. When you look at the United States, we know for sure that we are not the low-cost labor producer. We had better be the low-cost energy producer.

This means that it is essential that we have low-cost energy, dependable energy on into the future, if we are going to maintain our competitiveness in the world.

In addition to that, both my brother and I have lived and worked abroad, and we see the benefits of inexpensive reliable energy to lifting the lot of mankind in the Third World. We think that is very fundamental.

We think that if the Third World can be promised safe, dependable, low-cost power, it can mean progress. In fact, the progress of peoples all over the world can be pretty well traced to the amount of electrical energy that they have.

The reason for that is that virtually all production is powered by fractional horsepower electric motors. If you don't have power, the human being is left to his own strength, and that is not very much compared to what electrical can give.

The Third World must have the power that the developed parts of the world have enjoyed for many years. The alternative is a stagnation, a continuation of poverty, and that inevitably leads to political instability. And I think that is something that we don't want to wish on anyone.

When we look at the world markets, the HTGR, the modular HTGR has a couple of things that are rather unique to it. First of all, because of its passive features it is virtually idiot proof, and that is very helpful. Because around the world there are different levels of being able to cope with complex technical machines.

The other, of course, is the low-enriched, once-through fuel cycle that is particularly important for nonproliferation. So we see both from a domestic market and an international market that while in the United States there isn't a great need for energy today, there is certainly going to be if we are going to remain viable in world markets; and that, of course, we all know is essential.

Beyond that, perhaps businessmen have to understand technical problems in a very simple way. The modular HTGR has some distinct safety features which can be reduced to a rather simple form. I think there are five key points.

Of this five key points, three are absolute unique to the modular HTGR. Two are enjoyed by some of the other technical concepts. But real quickly, it starts with the fuel particles which are encased ceramics, and we all know how tough, noncorrosive that is.

Then you have the helium as the cooling medium, and it is an inert gas and it simply makes sense to have your cooling media inert so that if there is any kind of radiation release it is not carried and it doesn't get into the carrying medium. That just makes inherent sense, if you will.

The other factor is the graphite blocks. They are big. They are ponderous. They would hardly be something you would want to fuel an airplane with, but they make all kinds of sense for a reactor, because they are so stable. There is a low-powered density and it means that you don't have wide excursions, rapid excursions of temperature.

In fact, that is the key to the long time that you have to deal with any kind of problem that you have, because the graphite is so stable, you could go in and work on your problem over hours instead of, perhaps, the minutes or even seconds you have in some other, like they may have had a Chernobyl.

Of course the other two factors are the negative grade void coefficient, which some of the other technologies have; and the size. And the size is one of the very fortunate coincidences of technology. I think it is the most significant coincidence that has come along since the high-bypass-ratio fan engine on airplanes.

The objective when they developed the high-bypass-ratio fans was for fuel economy; and that did that in spades. It probably got 40 percent fuel economy. But one of the ancillary results was that they decreased by several magnitudes the noise emitted by jet aircraft; so today we have much quieter airplanes.

In case of the size, which a key to the safety of the modular HTGR—one of the keys, one of the five keys—you are able to take advantage of the factory production, quality control, all of the economies that come from that. You are able to take advantage of much smaller increments of predicting and ordering your future power requirements, 140 megawatts instead of 1,000 megawatts, and that is a big factor.

Because it is made in the factory and you can essentially order of the shelf, if you will. This means the time that it takes to do a plant is dramatically reduced. And this is terribly important in our regulatory environment that has sprung up as a result of Three Mile Island and Chernobyl. It is that time factor that has so aggravated so many of the nuclear plants.

The final factor, in terms of economy, is because when you maintain a 140-megawatt plant, instead of a 1,000-megawatt plant, and you put that reactor down, you are only cutting off maybe one-quarter, one-seventh, or one-tenth of your total capacity, and that means that the utility only has to plan for replacement of a small increment of the total of that reactor complex instead of the whole reactor; and those are big economic factors.

I would emphasize that these are factors that make sense to a businessman. We think they make a lot of sense technically. But it is the economics that ultimately are going to be terribly important to the utilities. We have worked hard to maintain close relations with the members of the utility committee, and we think that is another unique part of our technology. The utilities really are interested in this technology.

I would also add that the other key ingredient is the support that your committee has given this project. And we think that as we get into some of the technical details you will feel very gratified with what has taken place, the amount of progress that has been made.

I have with me today Gene Northup, who is general manager of the Power Reactor Programs at General Atomic. He will be able to give you a lot of the detail of what we have been doing.

But there is one other thing that I would leave with you, just really quickly. I heard Senator Hatfield talk about the importance of having plentiful energy in the world. He said that if there were ever another war—and perish the thought that there should be—but if there were, it wouldn't be because of communism and democracy, or socialism and capitalism, it would be because there was too much pressure on the world's energy resources.

Well, in furthering this technology, you are assuring that we won't get to a point where the world sees a volatile situation in terms of insufficient energy for the peoples of the world. I think that is a very important objective.

Frankly, it is exciting to GA to be a part of that. It is certainly exciting for my brother and me to be a part of it.

Thank you.

[The statement of Mr. Linden Blue follows:]

Subcommittee on Energy Research and Production
Committee on Science and Technology
United States House of Representatives

Status of the High Temperature Gas-Cooled Reactor Program

Linden S. Blue
Vice Chairman of the Board
General Atomic Technologies Corporation

Madame Chairman and Committee Members:

I am Linden S. Blue, Vice Chairman of the Board of General Atomic Technologies Corporation, the parent company of GA Technologies Inc., and I am most pleased for the opportunity of appearing before you today on behalf of the High Temperature Gas-Cooled Reactor program.

This past August 13, it was announced that General Atomic Technologies Corporation, a privately-held company owned by my brother, J. Neal Blue, and myself, had entered into an agreement with Chevron Corporation for the purchase of GA Technologies. In going forth with this major acquisition, Neal and I considered all of the aspects of GA -- the unique technological base which has been built up over the years; the broad applicability of the base to both near-term and future commercial and defense-related business; the record of achievement in areas such as research reactors, power reactors, fusion development, electronics hardware design and manufacture, and SDI systems; and the asset in uniquely talented people and the facilities. Of key importance in our decision was the opportunity that we saw in the Modular HTGR program and the belief that an energetic, entrepreneurial management can help bring this exciting concept forward where its benefits can be brought to both the domestic and international energy markets.

Neal and I are not nuclear specialists so our judgments and enthusiasm about the Modular HTGR are based, not on detailed technical considerations, but rather on a broader, macro view of the system, about nuclear energy, and about its role in the world energy picture. When we look at nuclear energy in the context of our backgrounds in business, energy and finance, we believe that the tide in nuclear has fully receded and that it will flow back

in. Oil and gas are currently plentiful and inexpensive. However, they are certain to grow scarcer and more costly. Excess generation capacity exists in the U.S. and many of the developed nations at this time, but this situation will reverse in the next decade, even with the conservative forecasts of energy growth. In the developing nations there exists an insatiable need for energy. Indeed, the third world must have dependable, low cost energy if it is to participate in the development and prosperity the western industrialized world enjoys. The alternative is poverty, political instability and, too frequently, violence. After considering both the advantages and disadvantages of the other alternative sources of energy, such as coal, it appears to us that nuclear energy is destined to play a major role in the future world energy picture, and that the Modular HTGR is the right concept to lead the industry to this future.

The Modular HTGR with its low enriched once through fuel cycle is suitable for export without increasing the risk of nuclear proliferation and I believe this consideration should be given to the development of all advanced nuclear systems. With the current trade deficit impacting our economy so severely, it is important that we not lose this opportunity to reestablish the U.S. as a major nuclear power exporter.

As you would anticipate based on the backgrounds of both Neal and myself, we are dedicated to developing the potential of the Modular HTGR for cooperative international participation so it can serve the vast international energy market with safe, economical nuclear power.

The Modular HTGR is an advanced second-generation reactor system which is able to benefit from over the 25 years of experience with gas-cooled reactor system design and operation throughout the world. As this committee is aware, the system is being specifically designed and developed to respond to the require-

ments and expectations which have evolved regarding the application of nuclear energy in today's society. These requirements take into account not only the altered energy growth rates in the U.S., but also consider the impact on the industry of the schedule and cost overruns, the changing regulatory requirments, and the impact on the public, regulators, and financial institutions of Three Mile Island and Chernobyl. We believe the small, efficient, economical, and inherently safe Modular HTGR is the right nuclear system and we intend to apply the resources of GA Technologies on advancing the development and commercialization of the system.

We recognize that this promising technology is very much dependent upon maintaining the support of the user industry and the Congress so as to allow this concept to be advanced to a point where it will be available for commercial deployment. I trust that the information that GA Technologies and the other participants in this hearing provide to you today will justify a continuation of the confidence and support in the program that your Committee has provided.

I have with me today, Mr. Gene Northup, who is General Manager of Power Reactor Programs at GA Technologies. This group is responsible for the technical and programmatic activity on the Modular HTGR program. I would like for Mr. Northup to review for you the current status and progress on the program elements being conducted by GA Technologies.

Again, I greatly appreciate the opportunity of appearing before this Committee whose support has been so vital in the direction and progress of this program.

I will be pleased to answer any questions that you may have at this time, or following the testimony of the other witnesses.

Mrs. LLOYD. Well, Linden, if I had not been sold already on MHTGR, I assure that I would be now. You are a good salesperson for this new reactor concept.

I agree with you, the time is now, not only for environmental reasons, for safety reasons, but for economic reasons that we need to move on with this.

One of the things I would like to point up that we discussed with Mr. Fawell earlier, that certainly most of our fossil plants are four or six increments that they do not depend on merely one system, but they are also, in fact, modular in concept.

But I want to go back on something that you said earlier. I think that if we can give the Third World countries the benefits of peaceful use of the atom, then there will be a less of a need, less of a reliance to rely on the military application of atomic power.

By using the atom for peaceful purposes, for industrial purposes, for the needed energy in their countries, that certainly we will give them a greater sense of security than any nuclear device could ever give them.

And I do applaud you.

What is you concept now of Third World countries, or our neighbors; your perception of the interest in MHTGR?

Mr. BLUE. Well, one of the important reasons why the Third World is important in the equation is because their energy need is now. Many countries have been through a very traumatic era that started in 1973 when suddenly either they didn't have energy, or they had to pay inordinate prices that virtually bankrupt them.

We don't want to return to that. We would like to see the benefits of nuclear power be extended to the Third World.

That market is open and available now if you you can have systems that are passively and inherently safe, and in small modules. There are many countries that simply can't speak for a 1,000-megawatt plant, who would be much more able to speak for smaller plants, in increments of 140 megawatts; that is No. 1.

No. 2, because that market is there now, I think it is quite possible that the first reactors to actually go into power production might be outside the United States. I think that would be a very good thing for the United States to be able to reestablish its preeminence in nuclear power with all that means for balance of payments.

So we would be very aggressive in developing international markets and we think that is a natural. The situation in this country, of course, the urgent need for more capacity is going to come in the mid-1990's. We expect to be ready for that.

But I think prior to that time there will be a vigorous international market for the right technology, and we think that is what we have.

Mrs. LLOYD. Do you feel that the Department of Energy is moving at a pace that makes sense; do you think the Department of Energy is dragging their heels with the MHTGR Program; or do you think we are on schedule?

Mr. BLUE. No; I think we are on schedule. I would be less than candid if I didn't admit that I am somewhat impatient sometimes and would like to see things move more rapidly. I want to see improvement in the world energy picture yesterday.

It is a complicated task, a complicated technology, and it does take time to develop properly.

I do believe that if there was more funding we could develop more rapidly; and frankly, that is one of things that we would like to do. But at least at the present levels we are able to do in a very systematic way, we are able to develop our technology.

We have been pleased with the support of the Department of Energy. We realize that there are other technologies that they are interested in. We simply feel that we have the best, particularly because of the fact that the customers want our technology. The Gas Cooled Reactor Associates represents 30 percent of the generating capacity in this country; that is pretty significant.

They are another good source of that.

Mrs. LLOYD. As you know the philosophy of this subcommittee is that our role is to develop technologies that the private sector can't do alone, and to develop the technologies where there is an actual need at the marketplace. We believe that this is the technology for today, it will answer not only our needs for environmental, safety and for economic reasons.

We thank you for what you are doing. We thank you for being a good salesman for the MHTGR.

Mr. BLUE. Because I believe very strongly in it.

Mrs. LLOYD. Thank you very much.

Mr. BLUE. Thank you.

Mrs. LLOYD. Excuse me, Mr. Blue, Mr. Fawell came in, pardon me.

Mr. FAWELL. I have no questions.

Mrs. LLOYD. Pardon me, I didn't mean to overlook you Mr. Fawell.

We have a panel of witnesses at this time.

Our next panel includes Mr. Gene Northup, from GA Technologies; Mr. Warren Chernock, from Combustion Engineering; Dr. Paul Kasten, who is from the Oak Ridge National Laboratory, and happens to be one of my constituents.

I am very proud of Dr. Kasten; he means a lot to our district. As well as Dr. James Kendall, from Gas Cooled Reactor Associates in San Diego.

Gentlemen, we are happy to have you here. We know we are going to benefit from your testimony.

Mr. Northup, you may go first.

We will go down the line and you may proceed as you wish and your entire written statement will be a part of the record.

STATEMENTS OF GENE NORTHUP, GENERAL MANAGER, POWER REACTOR PROGRAMS, GA TECHNOLOGIES; WARREN CHERNOCK, VICE PRESIDENT, R&D NUCLEAR POWER SYSTEMS, COMBUSTION ENGINEERING, INC., WINDSOR, CT; PAUL KASTEN, OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TN; AND JAMES KENDALL, DIRECTOR OF PROGRAM DEVELOPMENT, GAS COOLED REACTOR ASSOCIATES, SAN DIEGO, CA

Mr. NORTHUP. Thank you, Madam Chairman. I have submitted a written presentation, and I will excerpt from that very briefly to give you the highlights.

I am Gene Northup, general manager of GA Technologies, Inc., their Power Reactor Program group, who is responsible for the modular reactor development within our company.

It is a real pleasure for me to appear before this committee to report on what I think is a rapidly escalating activity on the design and the progress that they are making. I hope when I finish you will see how far we have come.

This program is well organized. We know where we are going. We know what we need to have done. And we know that the specific status that we will show you here—and I think we have given you handouts which show some of the detailed design drawings that we have this stage—will give you an idea of how excited, perhaps, I am, and if I show my excitement please forgive me, because——

Mrs. LLOYD. That is what you are here today for.

Mr. NORTHUP. A comprehensive design basis for the 350 megawatt modular HTGR has been established. All of the participants have concentrated in pulling this design together, so it is not just a one-organization show. It has been defined and we believe now that it can be defended.

A five-volume preliminary safety information document has been prepared for submittal to NRC at the end of this month, which is right on schedule from where we set out a year ago.

Due to the amount of design work that we have accomplished, we are now prepared to defend this submittal before Tom King and his group at NRC. And we are after this affirmative licensability statement and the safety evaluation report early in fiscal year 1988. We think a minimum of the $25 million is required to meet these objectives within NRC, but also to expand the design detail. All of the top design criteria documents and design descriptions have been issued. There are about 100 of them. They establish and will control the design to the end of the program; these will be our bibles to follow. These documents include approximately 110 major plant design drawings, as Dr. Wilcox said. Over 43 major component, top assembly, and detailed drawings supporting those have been prepared. The majority of these will be submitted in the PSID that goes before NRC. Examples of these have also been put in the handout that you have received. Over 200 computer codes have been adjusted and readied for the modular reactor. This is taking from use on large plants and making them applicable to this design. At least 20 of these have also been taken through national laboratories for independent review and for verification for use on the design, where there is also a peer review established.

A design control procedures are in place for the program, via the DOE's plant design control officer, in which all of the participants are involved. Dr. Wilcox mentioned that we use 19.8 percent enriched fuel. We use also thorium in the fuel cycle which is a conservation of the uranium. We have no reprocessing required. We meet all nonproliferation requirements as well as the economic requirements from this fuel that is put on us from the utilities.

The licensing plan has been submitted and approved by the NRC. We have had 12 topical meetings with their staff. And we have had two meetings with ACRS. One specifically at the full

committee's invitation. The next meeting with NRC staff is scheduled just in a couple of weeks, on September 30.

A detailed probabilistic risk assessment and regulatory technology development plan will be submitted the first part of 1987, just a couple of months. And numerous meetings on these are scheduled through 1987. It is our hope that NRC will be able to maintain the schedule of activity that we have laid out.

I think we have been very successful at this point. We have four working scale models that have been built, and these have been able to be produced this early in the schedule only because of the interfaces that we have defined and brought together. Normally models are made much later. One of the model pictures is in the front of the handout that we have also given to your committee. We have a new advent of the CAD/CAM computer which allows us now to do three-dimensional modeling in a computer form, and this is a tremendous acceleration tool in finding how things fit together or they don't fit together, and we can make corrections. You will also find in your handout those examples of those computer-generated, three-dimensional drawings for use on this design.

To date, the MHTGR design effort is showing that the goals set by this committee in November 1983, and that have been quantified by the utilities through GCRA, can meet the significant milestones set with margin. Specifically, in regards to passive safety, the heart of the MHTGR, as it was mentioned by Linden, is the fuel. Imagine, if you will, a large nuclear plant containment structure being shrunk down into size to just cover a piece of nuclear fuel the size of a poppy seed. If you will make one more mental picture and you take three of those and shrink three more, and shrink it down again around this particle, you will have what we mean by the coating around nuclear fuel. One more mental picture, and that is if you assume or visualize that this little shrunken vessel now will withstand the pressure of 50 containment structures, I would hope you would also carry this with you as a remembrance of what the fuel can do. And Mr. Fawell mentioned his concern of storage and radioactive waste, the fuel once it is in this particle can remain there for its also storage; we do not have to reconstitute it or put it into different packaging for its waste control.

I hope now that every time you see one of these little roles with poppy seeds on it you will think about this structural significance. I am a structural engineer, and I think it is terrific.

The goal of developing now the powerplant around this particle has been to set the system such they would never violate these particle coatings. And we agree that we have such a design, even to the point where the control room of the plant does not have to be safety grade. There is no safety grade requirements on the control of this system. The offsite doses are well below regulatory requirements, and evacuation or sheltering of anyone outside the plant boundary of 1,500 feet is just not required.

Natural convection cooling of the core without use of pumps or valves will be done. In fact, it has already been done on a test reactor, the AVR reactor in Germany, 15 years ago. Where the elements of the heat transfer were established in a test that was run on that reactor.

Now, how do we know the fuel is so good? There has been over 1 million of these particles that have been accelerated in the radiation tests, over a billion have been run in reactors, and we have examined many, many, of these particles in microscopic examination to establish their parameters. This provides an enormous data bank of structural information. And yes, these little shrunken vessels are designed, they are not just put in the uranium particle, there is a design process that goes with it. Specifically, we have produced 50 KG of high quality coated fuel particles for the MHTGR specifications, specifically, and these were produced this last year in a full-scale, pilot production segment at GA Technologies. So the beginning parts of the production line for making this fuel have now been demonstrated.

You mention the Chernobyl accident; and this certainly has drawn attention to reactors with graphite. We know that large amounts of air and very, very high temperatures are required to start let along sustain any oxidation of graphite. But here we let the laws of physics help us.

Due to the silo location—and if you will notice in the model picture in the front of your handout—this unit is completely below ground. The amount of air ingress is so limited that very little interaction could ever take place, even if a fuel element—this is a large graphite block with fuel in it—was taken out of the active core and placed on the ground; it will not burn; and there will be inappreciable amount of oxidation occurring.

On building quickly, it was mentioned that we wanted shop fabrication. Well, shop fabrication via these vessels and small modules is fully realizable—and Mr. Chernock, I am sure, will mention this in his discussion on fabrication—but the overall projections for an equilibrium plant show that the construction can be completed in 27 months from the start of site construction to the full power production.

On the availability, the concept will meet 80 percent availability which was the goal, in fact we have had the Small Peach Bottom reactor here in the United States and the AVR reactor reactor in Germany, which have exceeded this 80 percent with margin. The plant is designed, as I said, for decay heat removal by natural convection cooling even with the total loss of the coolant.

We can throw the coolant away and we are still able to cool a core. Now, as a result of this, the goal of preventing the loss the investment to the owners has been put in the category where they will not have to sustain this even in a very large population of plants of this similar type.

The MHGTR economy looks very good. It shows a significant power cost advantage, at least 20 percent over coal, for the same size of plant.

While something probably has been mentioned about the schedule, and although the current DOE schedule recognizes funding restrictions and the nuances of the political scene, we certainly believe that it could be accelerated because we are on a very well defined path. The development program does not require large test facilities or infrastructure.

Testing can be phased with the design effort. Present indications are that the overall cost for the design development and the ope

ation of the first demo is around $800 million total. And the first demo can be operational by the end of 1995, assuming that the funds that we have laid out are provided on a yearly basis. This would be confirmed by completing the preliminary work that we also have planned.

On the international cooperation it has already been mentioned how well that is going in the support of our program.

I would mention that as a result of some privately funded work with the Japan Atomic Energy Institute, that we believe they will select the United States prismatic block for their very high temperature reactor test program. That has only come about because of our ability to have a technology interaction with that group.

As has previously been indicated by Mr. Blue, GA Technologies is deeply committed to the success of this program. We are prepared to apply our energies and available resources to see the program move forward.

We do look toward the government participation and support in those areas of first of a kind, where the risk that cannot be prudently taken by the private sector are covered.

I trust is evident from my comments and the material that you have as handouts, that substantial progress has been made in a very short time, and that we are ready to move forward.

I thank you for the opportunity to appear before you. I hope that you will continue to show your enthusiasm for our program.

[The prepared statement of Mr. Northup and attachments follow:]

Statement Submitted for the Record

Subcommitee on Energy Research and Production

Committee on Science and Technology

United States House of Representatives

Modular High Temperature Gas-Cooled Reactor Program Review

T. E. Northup
General Manager
Power Reactor Programs
GA Technologies Inc.

September 17, 1986

Madam Chairman and the Committee Members:

As Mr. Blue just commented, I am T. E. (Gene) Northup, General Manager of GA Technologies Inc.'s Power Reactor Program group. It is a pleasure to appear before this Committee today to report on the rapidly escalating progress which is being made on the Modular High Temperature Gas-Cooled Reactor program. The technical concept and status are such that the Modular HTGR program is attracting significant notice in the media and in the energy community. We expect this program recognition to continue and expand.

I will review today some of the more significant accomplishments of the Modular HTGR program since Dr. Dean and Mr. Carey appeared before you last March to report on the selection of a single concept.

Status of the Modular HTGR Design

Following the concept selection of the 350 MW(t) Modular HTGR with an annular core of prismatic fuel in September 1985, all of the participants (DOE, GA Technologies, General Electric, Combustion Engineering, Bechtel National, Inc., Stone & Webster Corporation, Oak Ridge National Laboratories, EG&G, and the Gas-Cooled Reactor Associates) have concentrated on moving the design forward. Significant progress has been made in all areas of the plant and in particular on the conceptual design of the nuclear island.

The status of the design at the end of FY86 is the following:

Overall Plant Design Specification Issued
System Design Descriptions Issued, including
- Reactor System
- Heat Transport System
- Vessel System
- Reactor Cavity Cooling System
- Plant Protection and Instrumentation System
- Fuel Handling and Storage System

Subsystem Design Descriptions Issued for all Subsystems
Preliminary Safety Information Document Submitted
Technology Development Plans Issued for
- Graphite
- Fuel and Fission Products
- Metals

Regulatory Technology Development Plan Drafted
Probabilistic Risk Assessment Drafted

The current MHTGR design embodies new and innovative features, it has evolved in an evolutionary rather than revolutionary manner and embodies the following: 1) experience from operating HTGRs (Peach Bottom 1, Fort St. Vrain, and AVR) 2) technology from LWRs (e.g., steel reactor vessels) and 3) features compatible with passive safety characteristics.

As with all high technology endeavors, several iterations on the design were necessary to meet all the requirements, and in many cases this involved performing trade studies to identify the optimum and cost-effective approach.

The plant design established at the end of FY '86 embodies the results from many NSSS and BOP trade studies, and results from engineering analysis of various accident scenarios prepared to support the PSID, and also provides a better definition of economics. While the current status is still classed as conceptual, many projects have in fact moved into the preliminary design phase with less technical definition.

Advancements planned to complete the scheduled design in FY '87 include: 1) rationalization of issues stemming from review of the PSID by the NRC to permit an affirmative licensing statement to be issued, and 2) a definitive cost estimate and schedule. It is our estimate that this work can be completed quickly, subject only to the level of funds provided in FY '87.

As further description of the current status of the design, I would note that the design has already reached the stage where four working scale models have been produced to uncover and eliminate major interface conflicts, evaluated construction sequences, and study operation and maintenance needs.

Numerous computer generated 3-D solid models of the nuclear portion of the plant have been prepared to make sure the concept meets arrangement and access requirements. Examples are in the separate handout.

During FY '85, approximately $30 million was applied to the selection of a supportable concept and during FY '86 approximately $26 million was applied to its design detail to produce a Preliminary Safety Information Document for NRC to review.

National Laboratory support has been at the 3 million dollar level with efforts concentrated on reactor shielding, design review, fuel development and graphite and material technology. Contractor support has been at the 23 million level with scopes assigned as follows:

GA Technologies:	plant and nuclear steam supply design, licensing and assessments, fuel process technology, and graphite and component technology
Combustion Engineering:	vessel and steam generator design and supporting technology
Bechtel National, Inc:	nuclear island design, radiological doses and constructability
Stone & Webster Engineering Corp:	energy conversion area design, plant cost, and construction schedule
General Electric:	plant control, simulator, operation and maintenance, and advanced materials

Dr. Wilcox's testimony covered many of the key accomplishments and the following should provide you with specific details and insight into the completed work.

All of the top design criteria documents and design descriptions have been issued. They include the Utility User Requirements, the Overall Plant Design Specification, 8 System Descriptions, 32 Subsystem Descriptions, 15 Interface Control

Documents and 39 design reports. These documents include approximately 110 design drawings of the plant covering structural, electrical, piping, instrumentation, and controls. Over 43 component top assembly and detail design drawings (many have multiple sheets) were issued for vessels, steam generators, heat exchangers, refueling machine, circulators, core internals, and control rod drives. The majority of these will be in the PSID being submitted at the end of September to NRC. Examples of some of the top assembly drawings have been included in the DOE handout.

At least 200 computer codes have been adjusted or upgraded for direct use on the MHTGR. At least 20 major computer codes have been independently reviewed and verified by national laboratories to provide a sound reference base for proceeding with preliminary and final design. For example, MLTAP, PANTHER and TAC programs have been used to perform steady-state and transient analyses, including core conduction events. Detailed component analyses have been performed for the steam generator using the NUSIZE and SUPERHEAT codes, and for the reactor core using the SNIFFS, TAC and HEAT2 codes. Design control procedures are in place to assure appropriate development of the design and licensing process.

All of these documents and code development work have been essential to the completion of the Preliminary Safety Information Document (PSID) which will be submitted to the NRC September 30, 1986 for review and approval during FY87. In support of this PSID submittal, the core physics design has been completed through the conceptual design level and well into the preliminary design phase. There are no feasibility issues related to the core design; however, the detailed design work is required to bring the thermal-hydraulic and mechanical design to the same level. The work that has been completed includes the specification of the fuel loading,

enrichment, and in-core distribution, for every cycle. Many fuel cycles are possible in the MHTGR core depending on economic conditions, but the reference cycle uses Uranium enriched to 19.8%, plus Thorium as a fertile material, with no reprocessing.

Licensing Activities with NRC and ACRS

A Licensing Plan has been submitted and approved by NRC which lays out the schedule for detailed interaction with the staff. Although budget adjustments during FY'86 caused NRC to curtail its interaction with the MHTGR program for about six months, twelve topical meetings with NRC staff, and two meetings with ACRS were held. Specific licensing information which has been submitted to the NRC and briefing meetings are shown in Attachment 1 to this Testimony. Future interaction planned is in Attachment 2. NRC has responded favorably to the MHTGR Top Level Regulatory Criteria prepared by the participants and submitted by DOE and has suggested that all advance reactors use the same approach. NRC has recently reinstated HTGR funds and the next meeting with NRC staff is scheduled for September 30. A key event, which is proceeding on schedule, is the submittal of the Preliminary Safety Information Document in September. This 5 volume report describes the design, the application of the licensing bases, and assesses the plant's response to various accident scenarios and its ability to meet top level criteria. This document will be supported by a detailed Probabilistic Risk Assessment and Regulatory Technology Development Plan to be submitted in the first quarter of 1987. Numerous meetings are already scheduled for FY'87. Initial staff briefings will cover issues identified by the NRC. These include standardization, accident selection/source terms, and containment/confinement. These briefings should help clarify NRC concerns and either provide sufficient information for resolution or guide the development of further data.

It is our hope that NRC will maintain the schedule of activity as planned and achieve the goal of an affirmative licensability statement and Safety Evaluation Report in early FY '88.

Are Goals Set Being Met?

All the paper produced is just that unless the MHTGR being developed can meet the goals set forth by your Committee in the Fall of 1983, and as expanded further by the utility users represented by GCRA.

- Passive safety
- No evacuation or sheltering of the public
- Build quickly - shop fabrication
- High availability - (greater than 80%)
- Investment protection
- Economical - (10% less than coal)

These or very similar goals have been adopted now by all advanced reactor development groups. To date, the MHTGR conceptual design effort is showing that the goals can be met with significant margin.

Passive Safety - The MHTGR is inherently safe by incorporation of passive features to the point where even the control room does not have to be safety grade and the offsite doses are well below regulatory requirements at the site boundary such that evacuation or sheltering of anyone beyond 1500 feet is not necessary. A high statistical confidence in the triso-coated fuel coatings to retain fission products at the required level is provided by the fact that extremely large numbers of triso-coated particles have been tested. For example, several million coated particles have been tested in

accelerated irradiation tests and over a billion have been irradiated in FSV and AVR which is providing an enormous data bank for particle design and coating performance for use on the MHTGR.

The no-evacuation goal for the MHTGR, requires an improvement in the fuel coating quality and performance, and demonstration of the improved fuel manufacturing process. During FY86, coated particles meeting required quality were produced at a capacity of 2 Kg/day on a large scale pilot production line.

With the combination of extensive laboratory testing, modeling, and model proof testing, the oxidation of graphite by water vapor and the effects of oxidation is predictable - and endothermic. Based on FSV experience, in which moisture ingress has been uncommonly frequent, there has been no discernable effect on fuel elements or fuel observed through surveillance of removed items. The MHTGR is being designed such that even with large amounts of water ingress, such as might occur with failed S.G. tubes, there is no significant oxidation damage and the small amount of weight loss does not effect the strength of the core and core support components. The reactor can be cooled even with total loss of all coolant, not just on loss of flow. There are no pumps or valves to operate or fail in the natural convection cooling of the reactor core -- a core which cannot melt or interact with its helium coolant. These thermal calculations have been independently corroborated by national labs and other participants, both here in the U.S. and in other countries. The computer models and data used for these assessments include well established methods and data supported by extensive research and in some cases operating experience.

Experimental tests in operating reactors (FSV in the U.S. and AVR in Germany) confirmed the fundamental thermal-hydraulic mechanisms currently modeled. A scale model of the specific configuration of the MHTGR NSSS is planned to confirm the passive heat removal capability.

The Chernobyl accident has drawn attention to reactors using graphite. A detailed discussion on the merits of graphite in an HTGR is attached to this testimony (Attachment 3) and shows why graphite was of minor concern in the Chernobyl accident. A further comment on this issue is that high temperature research has shown retention of fuel and fission products by the refractory ceramic coating of HTGR fuel occurs at temperatures much higher than those associated with the failure of fuel cladding at Chernobyl.

A number of U.S. and European laboratory experiments have shown that the strength of graphite increases with temperature and neutron irradiation, up to temperatures and fluence levels well beyond those experienced in an operating MHTGR. Other effects of the operating environment are small dimensional changes as well as changes to the mechanical and physical properties. These have been well characterized. The response of the graphite to the operating environment as predicted from the laboratory experiments has been verified in operating reactors, Peach Bottom and FSV in the U.S. and other graphite moderated reactors abroad. The operating experience has confirmed the structural adequacy of graphite.

The ingress of air into the MHTGR has been assessed. Due to the silo location of the module, the amount of air ingressed through any assumed opening is small because it is dependent solely on diffusion against the flow of He and/or reaction products which produce a net out-flow.

A probabilistic risk assessment has been completed for submittal to NRC in the first quarter of FY'87 which identifies all limiting design basis events, none of which compromises the inherent safety of the MHTGR. In fact, the emergency planning zone (EPZ) need not be larger than the plant's own exclusion area boundary i.e., 1500 feet.

Build Quickly - Shop Fabrication - The major NSS vessels are within conventional nuclear vessel shipment experience and the internal components are well suited for shop fabrication and assembly into modules. Projections for equilibrium plant deployment show only 27 months required from start of site work to first unit power production. Plant construction and improved quality assurance will result from the replication of shop fabricated items and the great reduction of site work.

High Availability - Detailed O&M studies of design continue to support assessments that the concept will meet the 80% capacity factor goal set by GCRA in its user requirements. The utilization of a special shutdown cooling system enhances the availability projections for this plant.

Strong emphasis has been and will continue to be placed on design specifications to reduce maintenance complexity. Availability goals have been assigned and each system/component is judged against its goal as the design proceeds. The plant controls are being designed to minimize the need for operator interaction, even with the operation of multiple modules.

Investment Protection - Since the plant is designed for decay heat removal with natural convection, the complete loss of all active cooling does not jeopardize plant equipment and the goal of preventing the loss of the plant investment in the lifetime of a large population of similar plants is an attainable goal.

Economy - As has been reported by the Deputy Secretary, the power cost estimate for the MHTGR looks good. The MHTGR shows a significant power cost advantage (almost 20%) over coal for the same size of plant. The favorable economics result principally from the feature of passive safety which allows the overall simplification of the plant through elimination of most engineered safety systems, and allows the balance of plant to be designed and constructed to conventional (non-nuclear) standards.

DOE Program Schedule

The schedule for the MHTGR development is currently dictated by funding constraints and not technology development. Although there are no fundamental aspects requiring resolution before proceeding with preliminary and final design, special effects tests of various systems and components are prudent to verify new designs. The technology development listed by Dr. Wilcox, including the fuel quality improvements, helium circulator development, and heat removal mechanism testing can be paced with the

design effort. The development program does not require large test facilities or infrastructure. The following safety related technology and component tasks are planned:

1. Development of MHTGR fuel manufacturing processes

2. Qualification of MHTGR fuel through quantification of fission product release and transport characteristics

3. Characterization of graphite materials for use in graphite fuel elements and graphite internal structures

4. Determination of reactor vessel irradiation and elevated temperature materials data

5. Validation testing of component/subcomponent assemblies:

   - core support structures
   - neutron control assemblies
   - reactor cavity cooling system
   - plant protection and instrumentation equipment

Due to the hiatus in the NRC effort in FY86, the licensing review of the modular HTGR by the NRC has not yet progressed to the stage of identifying research to be performed by the NRC itself.

The majority of the base technology will be performed by the Oak Ridge National Laboratory supported by GA Technologies. Component tests will be performed in the U.S. according to available test facilities and expertise, with a selection still to be made.

Although the current DOE schedule recognizes funding restrictions, acceleration would be very possible. Potential schedule and cost impediments to completing the design and licensing center on maintaining a stable program with required funding and timely responses - particularly from NRC.

The schedule for design of the MHTGR is as follows:

- Preliminary Safety Information Document	9/86
- Complete preliminary design	9/89
- Complete final design	9/93

Supporting the design and licensing schedule are the following special milestones:

- Preliminary Safety Information Document	9/86
- Regulatory Technology Development Plan	5/87
- NRC review of PSID complete	9/87
- NRC issue Safety Evaluation Report	1/88
- Preliminary Safety Analysis Report	9/89
- NRC issue Preliminary Design Approval	9/91
- Completion of Technology Program	9/92
- Final Safety Analysis Report	9/93
- NRC issue Final Design Approval	9/95

The goals of the FY87 program are to support NRC review leading to a licensability statement for the MHTGR concept. In addition, progress on supporting technology programs - especially fuel process development - is planned. The scope of work includes development of design in increasing detail.

Work will be performed by the participants listed earlier.

A minimum of $25M is required to meet the scheduled objectives for FY '87.

Projected Cost and Schedule for a Demonstration Plant

A detailed program cost estimate and schedule will be made in FY87. However, present indications are that the overall cost for the design, development, and operation of a single demo plant is about $800 million and the first demo can be operational by the end of 1995 assuming no funding constraints.

International Cooperation

Cooperative programs on technology development for HTGRs are on-going with the Federal Republic of Germany and Japan. Areas of exchange with FRG are for Fuel Development. Graphite Development, Fission Product Transport Technology, Metals Development and Reactor Safety Studies. Areas of exchange with Japan are for Fuel Irradiation Testing, Graphite Technology and In-Core Instrumentation Development. The direct benefit of the FRG exchange since its start in 1978 has been valued at more than $10M to the U.S. program. The direct costs by the FRG in performing tasks for the U.S. is about $4M. The FRG plans for further design development and testing of fuel from 1981 through 1992 at a cost of $60M.

The Japan/U.S. exchange is more recent, having been signed in 1986.

In a separate activity, GA Technologies earlier this year completed under private funding a special design study of the core and internals for a Very High Temperature Test Reactor (VHTR) for the Japanese Atomic Energy Research Institute utilizing Fort St. Vrain type fuel. As a result of this cooperative work, it appears that the JAERI group will select the U.S. prismatic block type fuel element for their program. Additional interchanges are being projected. This has demonstrated the versatility of the MHTGR as a basic power source for use on even more advanced concepts for process heat and direct cycle applications.

Support by Utilities and Vendors

The utilities, as is being reported by GCRA, continue to support the MHTGR development and have continued in a parallel effort during FY'86 to define the overall plans and requirements for a demonstration project and the cost/risk sharing between public and private sector. Phase II of this effort is nearing completion. Our participation in both Phase I and II have included cost sharing of the work performed to date.

As has been previously indicated by Mr. Blue, GA Technologies is deeply committed to the success of the MHTGR program and we are prepared to apply our resources to see the program move forward. We do look toward the government participation and support of those areas of first-of-a-kind costs which cannot be prudently covered by the private sector.

I trust it is evident from the testimony of the DOE and the material accompanying this testimony, not just my comments, that substantial progress has been made since the program has focused on the MHTGR system.

Thank you for the opportunity to present the above status of the MHTGR design before your Committee.

ATTACHMENT 1

Licensing information submitted to the NRC

1	Licensing Plan	4/85
2	Top Level Regulatory Criteria	8/85
3	Preliminary Safety Information Document Outline	10/85
4	Bridging Methods	7/86
5	Accident Selection Criteria	7/86
6	Licensing Basis Events	7/86
7	Preliminary Safety Information Document	9/86

Briefings given to the NRC staff and ACRS are:

1	Modular HTGR	6/84
2	HTGR Technology	8/84
3	No Evacuation Criteria (NRC/FEMA)	11/84
4	Top Level Regulatory Criteria	1/85
5	Bridging Methods	1/85
6	Accident Selection Criteria	5/85
7	Functional Analysis	7/85
8	Concept Selection	7/85
9	Safety-Related SSCs	10/85
10	Principal Design Criteria	10/85
11	Reference Design	12/85
12	ACRS Subcommittee	1/86
13	Full ACRS	3/86
14	Standardization	9/86

ATTACHMENT 2

Licensing information being prepared for the NRC

1	Principal Design Criteria	10/86
2	Resubmittal of Top Level Regulatory Criteria	10/86
3	Regulatory Technology Development Program Document (RTDP)	12/86
4	Probabilistic Risk Assessment	12/86
5	Basis for Emergency Planning	2/87
6	Update to RTDP	5/87

Briefings presently scheduled or planned are:

-	Standardization	9/86
-	Commissioners	10/86
-	Accidents/Source Term	11/86
-	Containment/Confinement	11/86

ATTACHMENT 3

THE MHTGR AND CHERNOBYL

Graphite has long been considered a means for improving reactor performance and safety because of its excellent nuclear and heat transfer properties, its stability at very high temperatures, and its capability to provide a large heat storage capacity. The Modular High Temperature Gas-Cooled Reactor (MHTGR) is designed to take advantage of these graphite properties to attain an inherent and passively safe system.

The Chernobyl nuclear disaster and the subsequent fire has raised questions regarding the design similarities and differences between the Chernobyl (RBMK type) Boiling Water Graphite Moderated Reactor, the MHTGR, and other reactor systems. In particular, attention has been drawn to the use of graphite as a moderator.

Study of the Chernobyl accident indicates that graphite was not the cause of the accident. Further, that even if graphite had not been used as a moderator on that reactor, the accident, explosion, and release of radioctivity would still have occurred.

Chernobyl Reactor

The Chernobyl RBMK reactor is a pressure-tube type of reactor using zirconium alloy clad uranium oxide pellets as fuel, boiling light water as the coolant, and graphite as the moderator. The fuel assemblies consist of a cluster of 18 clad fuel pins located in vertical zirconium alloy process tubes through which the cooling water is circulated. The fuel assemblies and coolant temperatures and pressures are similar to those for commercial boiling water reactors. The approximately 1700 process tubes which serve as the pressure boundary for the system pass through stacked graphite moderator blocks. The reactor is surrounded by concrete and steel shielding and support structures.

ATTACHMENT 3-2

Modular HTGR Reactor

The HTGR uses coated particle fuel, helium gas as a coolant, and graphite as the moderator and core structural material. The fuel particles consist of microspheres of uranium oxy carbide clad with pyrolytic carbon and silicon carbide, which form a barrier and pressure vessel to retain fission products. These refractory coatings are stable to very high temperatures. The coated particles are bonded together with a graphitic binder and inserted into large graphite fuel element blocks. The helium gas passes through coolant holes in the fuel element blocks. The pressure boundary for the MHTGR system is a steel reactor vessel which is contained in a below ground silo.

Different Features Affect Safety

The significant differences between RBMK and HTGR reactors are reflected in their susceptibility and response to accidents.

Fuel Heatup Rate on Loss of Coolant

- Slow for the MHTGR since it is determined by the heat capacity of the massive graphite moderator to which it is physically coupled.

- Rapid for the RBMK since the heat capacity of the fuel assemblies is small and the fuel assemblies are thermally isolated from the graphite moderator.

Temperature Stability and Margin

- The refractory MHTGR fuel particle coatings maintain their integrity to retain fission products to temperatures of over 3200°F. The graphite core structure gains in strength to 4500°F and is structurally sound to 5400°F.

ATTACHMENT 3-3

- The zirconium alloy RBMK fuel cladding loses strength and starts release of fission products at approximately 1300°F.

Coolant Reactions

- The MHTGR helium coolant is inert and does not react with graphite, fuel coatings, or other system components.

- In the RBMK, an exothermic (heat producing) chemical reaction between the water coolant and the zirconium alloy fuel cladding is initiated at about 1800°F.

Graphite Burning

It is difficult to induce self-sustained burning of the large, dense graphite shapes used in either reactor. Under ideal controlled conditions, graphite temperatures in excess of 1300°F are required. Such conditions could result only as a consequence of a major accident which could both raise the temperature of the graphite moderator and cause multiple breaches of the barriers normally preventing air contact with the graphite. The consequences from such an accident are limited for the MHTGR.

- The below grade level silo installation limits the quantity of air available to sustain combustion.

- The silicon carbide fuel particle coating retains its capability to retain fission products, even if the graphite has burned away.

ATTACHMENT 3-4

Reactivity Accidents

The nuclear design of the Chernobyl core makes the reactor power level difficult to control under certain conditions. An accident leading to a loss of the water coolant and/or an increase in the graphite temperature can cause a very rapid increase in reactor power, to steam formation, and to overpressurization and failure of the system. In the MHTGR, the nuclear reaction is unaffected by the loss of helium coolant and increased fuel or graphite temperatures result in reduced power outputs and ultimately to the reactor shutting itself down.

- The MHTGR has a negative temperature and power coefficients which protect the reactor against rapid power excursions.

Wigner Energy Release

A buildup of radiation induced (Wigner) energy in graphite can lead to a rapid increase in temperature when released. This is not a consideration in the HTGR or RBMK reactor, since both reactors operate at temperatures at which the energy is annealed out of the graphite continuously.

Water-Graphite Reactions

In either the RBMK or MHTGR reactor systems, water and graphite come in contact only as a result of equipment failures or accidents which breach a portion of the primary pressure boundary. At above normal operating temperatures, water and graphite reactions can lead to the formation of hydrogen and carbon monoxide; however, the reaction is strongly endothermic (absorbs heat) and is self-limiting.

ATTACHMENT 3-5

Loss of Cooling or Coolant Accidents

The nuclear accidents of major concern are those that can result in releases of fission products from the fuel and the failure of the pressure boundary allowing the radioactive material to leave the system environment. These accidents would characteristically involve a loss of cooling or of the coolant.

In the RBMK, coolant system blockages or breaks are conditions which could lead to a loss of coolant accidents. If immediate corrective action is not taken, failure of the zirconium alloy clad and the release of fission products into the system can take place in a matter of minutes. Further damage to the cooling system and/or exothermic chemical reactions (such as the zirconium water reaction) could occur which could lead to a breach of the primary pressure boundary and to radiation release from the plant. For the MHTGR, the loss of cooling involves the loss of helium coolant pressure, along with the loss of all active means of circulating the coolant or of removing heat from the core. In the MHTGR the fuel temperature increases very slowly so sufficient heat can be removed from the core by passive radiation and conduction to the surroundings such that temperatures never reach levels at which the refractory fuel cladding will fail or release fission products within the system.

Inherent and Passive Safety

The passive safety inherent in the fuel system, coolant, and plant arrangement for the MHTGR, which are dramatically different from the RBMK type system, essentially preclude the possibility of an accident such as that reported at Chernobyl and, more importantly, prevent accident consequences which would result in the significant release of radioactivity from the plant or for the need of public evacuation.

The Modular High Temperature Gas-Cooled Reactor

The Modular High Temperature Gas-Cooled Reactor

Modular High Temperature Gas-Cooled Reactor (MHTGR) is a second generation nuclear power system which can satisfy the concerns of the public, the government, the utilities and the investor community about nuclear safety and investment protection. Based on technology developed and demonstrated in the U.S. and Germany, the unique system makes use of refractory coated nuclear fuel, helium gas as an inert coolant and graphite as a stable core structural material. The safety and protection of the plant investment is provided by inherent and passive features and is not dependent upon operator actions or the activation of engineered systems. The high performance MHTGR provides flexibility in power output and siting, competitive energy costs, and can serve diverse energy needs both domestically and internationally. The design of this inherently safe MHTGR is being led by GA Technologies as a part of a cooperative program with other industrial contractors, the U.S. Department of Energy and electric utilities.

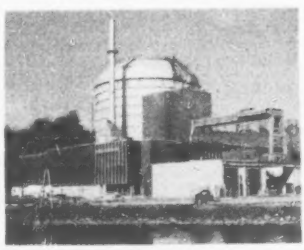
An Inherently Safe Nuclear Power Option

The Modular High Temperature Gas-Cooled Reactor (MHTGR) is being developed in the United States under a cooperative program involving the U.S. Government's Department of Energy (DOE), the nuclear reactor industry led by GA Technologies Inc. (GA), the industrial pioneer of the HTGR concept, and the utility/user industry represented by Gas-Cooled Reactor Associates (GCRA). The design is an advanced version of the basic High Temperature Gas-Cooled Reactor (HTGR) concept such as the Peach Bottom No. 1 reactor and the Fort St. Vrain reactor in the U.S. and the AVR and THTR reactors in the Federal Republic of Germany. Common to all HTGR systems is the use of:

- refractory coated particle fuel capable of retaining fission products at very high temperatures;
- graphite moderator which remains stable to very high temperatures and has a high heat capacity; and
- helium coolant which is inert, non-corrosive and remains as a gas under all operating conditions.

HTGR systems can operate at high temperatures to provide high efficiency electricity generation and the potential to access the cogeneration and high temperature process heat markets. The basic characteristics also provide inherent advantages associated with safety, investment protection, and environmental compatibility which are greatly enhanced by the special design features of the MHTGR reactors.

The low thermal output, the annular geometry of the core, and the installation of the reactor system in a below grade silo are design features incorporated in the MHTGR to provide safety which is not dependent on active safety systems, on operator actions or on the evacuation or sheltering of the public. The MHTGR provides inherent safety.



Peach Bottom No. 1 Reactor

Program Background – An Established Technology

The MHTGR design is based on a substantial background of gas cooled reactor technology experience gained over the past thirty years. The gas cooled reactor concept originated in the United Kingdom in 1956 with the Magnox class power reactors. These systems used carbon dioxide as a coolant and natural uranium as a fuel. The Advanced Gas Cooled Reactor is an evolutionary extension of the Magnox reactor that raises core outlet gas temperatures to 1200°F and modern steam systems. The nearly 700 reactor years of operating experience with gas cooled reactors in the United Kingdom form a key element of the technology base.

Modular HTGR Meets User Requirements

- Plant sizes 150-600 MW(e) range
- Equivalent availability >80%
- Probability of reactor module loss $<10^{-5}$/module-yr
- Meet existing safety and licensing criteria with no public sheltering for accidents with frequencies $>5\times10^{-7}$/year
- 10% power cost advantage over coal

The 330MW(e) Fort St. Vrain Plant

United States

GA initiated the development of the HTGR in the United States in 1957. The initial HTGR project in the U.S. was the 40 MW(e) Peach Bottom No. 1 developmental plant that was on the system of the Philadelphia Electric Company. This plant, built as part of the U.S. Atomic Energy Commission's Power Reactor Demonstration Program, operated very successfully from June 1967 until completion of the planned testing program and its scheduled shutdown in 1974.

The Fort St. Vrain 330 MW(e) demonstration plant project was also a part of the Power Reactor Demonstration Program. The plant, designed and built by GA for Public Service Company of Colorado, went critical in January 1974 and, through the end of 1985, had produced over 3.9 billion kilowatt hours of electrical energy. This first-of-a-kind facility has provided important confirmation and demonstration of specific design features and inherent operating characteristics of HTGR systems and equipment including the prismatic fuel elements, once through steam generators, axial flow circulators and fuel handling equipment.

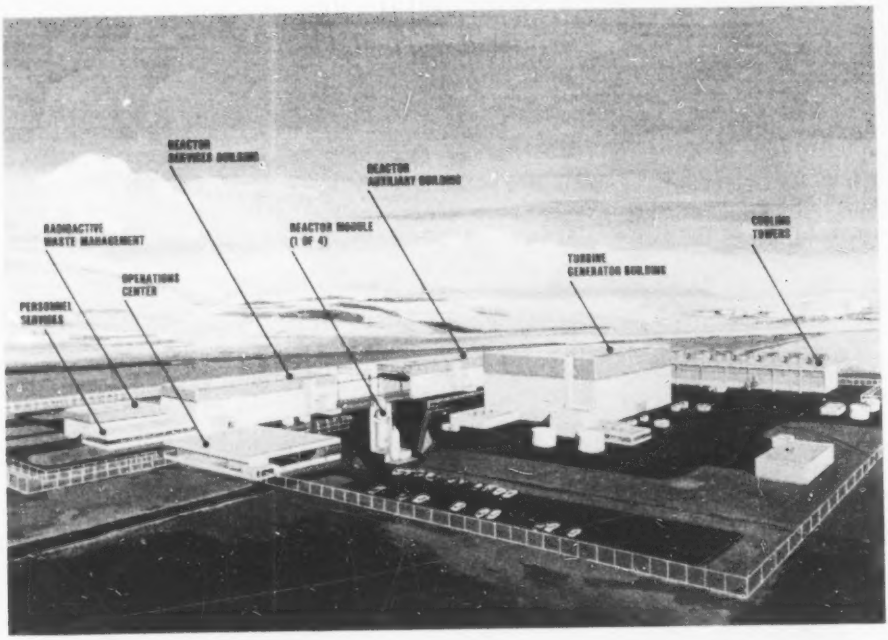
Germany and Japan

The HTGR program in the Federal Republic of Germany parallels the U.S. program and builds on the very successful 15 MW(e) Arbeitsgemeinschaft Versuch Reaktor (AVR) development plant which has been operating very successfully since 1968. The German program is currently in the process of starting up the 300 MW(e) Thorium Hochtemperatur Reaktor (THTR) demonstration plant. Both the AVR and THTR reactors use a pebble bed core design. The significant Japanese HTGR program has been pursuing the development of a Very High Temperature Gas Cooled Reactor.

To date, approximately $3 billion has been spent worldwide to develop specific HTGR technology. In the U.S., about $1.4 billion has been spent, of which over 60% has been provided by the private sector.

Matched to a Changed Energy Environment

Until 1984, the emphasis in the U.S. HTGR program was on the design of large single reactor power plants with outputs up to 1200 MW(e). Recognition by the U.S. Congress, the DOE and industry program participants of significant changes in the environment for nuclear power, including the financial, economic, and political considerations, coupled with distinct changes in the load growth pattern led to a reevaluation of the program. The evaluation by the joint industry/government participants led to the consensus decision to proceed with the development of the smaller MHTGR power plants, with emphasis on improved safety, greater reliability, and competitive economics. In mid-1984, the U.S. HTGR program effort was shifted to this effort and definitive user requirements were established to guide the design effort.



MHTGR Plant

The MHTGR plant design consists of four reactor modules, each rated at 350 MW(t), coupled to two steam turbine generators yielding a net power output of about 550 MW(e). The modular concept permits the flexibility of sequential deployment of individual reactor modules to match the end user's load growth requirements or financing constraints.

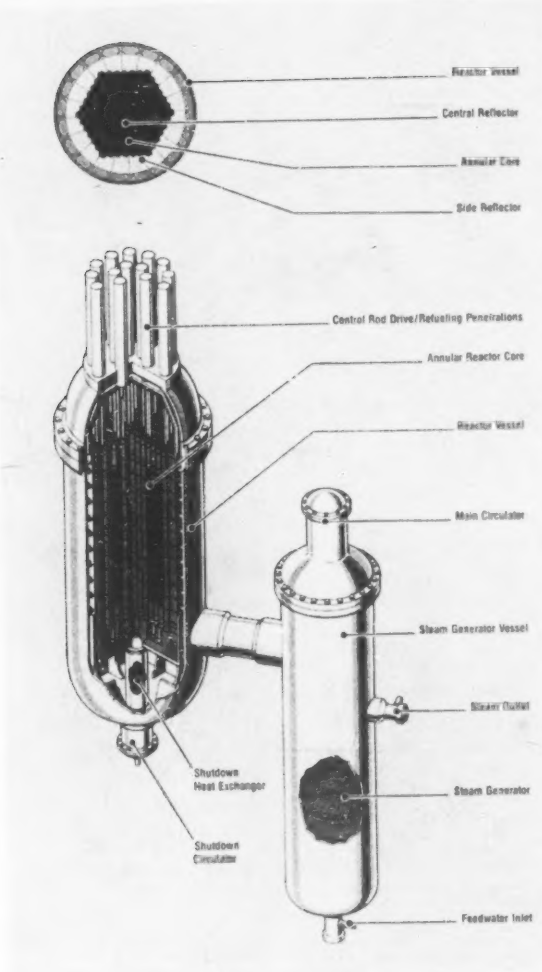
The four-module plant is divided into two major areas: the Nuclear Island comprising the reactor enclosures, reactor modules, and the safety related systems; and the balance of the plant housing the electric power generating systems. Each reactor module is housed in a vertical cylindrical concrete enclosure which is fully embedded in the earth. Each reactor enclosure serves as an independent containment structure having a vented and filtered exhaust system. The Nuclear Island portion of the plant also includes auxiliary structures which house common systems for fuel handling, helium processing, and other essential services.

Reactor System

The modular reactor components are contained within three steel vessels: a reactor vessel, a steam generator/circulator vessel, and a connecting concentric cross-duct vessel. The reactor vessel, 22.5 ft. in diameter and 72 ft. in length, contains the core, reflector, and associated supports. A shutdown heat exchanger and a shutdown cooling circulator are located at the bottom of the reactor vessel. Top-mounted standpipes contain the control rod drive mechanisms and hoppers containing boron carbide pellets for reserve shutdown. These standpipes are also the access ports for refueling and in-reactor inspection.

MHTGR Plant Summary Design Parameters

Thermal Power	1400 MW(t)
Electric Output	550 MW(e)
Net Efficiency	39.3%
Steam Conditions	1005°F, 2515 psia
Helium Temp. Core Outlet/Inlet	1268/487°F
Helium Pressure	925 psig
Core Power Density	5.9 W/cm³



MHTGR NUCLEAR STEAM SUPPLY SYSTEM

Reactor System Components

Steam Generator

A helical coil steam generator and the electric motor driven main helium circulator are contained in the steam generator vessel, which is 14 ft. in diameter and 85 ft. in length. Feedwater enters the steam generator through a bottom mounted header and superheated steam exits through a side mounted nozzle. The main helium circulator is mounted on the top of the steam generator vessel.

Primary Circuit

In the primary flow circuit, the helium coolant flows downward through the core where it is heated by the nuclear reactions. The hot helium then flows through the inner cross-duct and downward over the steam generator bundle where its heat is transferred to the water to make steam. The cooled gas then flows upward in an annulus between the steam generator bundle and the vessel, is recompressed by the circulator and driven into the annulus between the inner and outer cross-duct. The cool gas entering the reactor vessel flows up through the annulus between the core and the reactor to the top of the core to complete the circuit.

Secondary Circuit

In the secondary flow circuit, feedwater is transformed to superheated steam in the once through, spiral wound steam generator. After exiting the steam generator vessel, the steam is piped to a two turbine generator unit. Exhaust steam from the turbine is condensed and returned to the steam generator through a standard condensate/feedwater system.

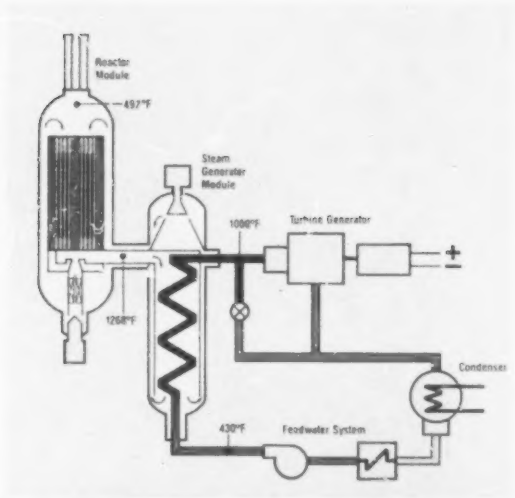
Reactor Core

The active core region is composed of graphite fuel element blocks that are hexagonal in cross section, 14 in. across the flats and 31 in. long. The fuel is in the form

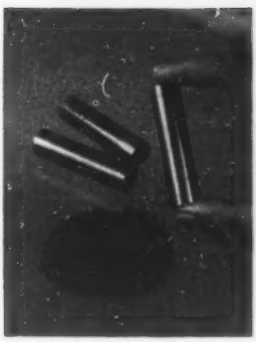
Control Rods

Decay Heat Removal

Refueling



SCHEMATIC FLOW DIAGRAM



Refractory coated fuel particles with fuel rods containing fuel particles.

Graphite fuel element containing fuel rods.

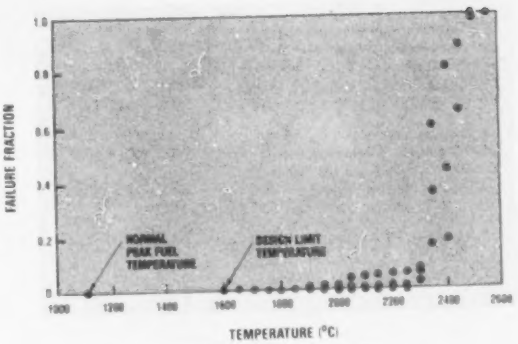
Control

Control of operations for the complete plant is performed from a single control room in the control building located adjacent to one end of the reactor services building.

Inherent and Passive Safety – A Primary Consideration

Inherent Features

The safety characteristics of the MHTGR are a result of the ability to effectively utilize the inert coolant, the capability of the refractory coated fuel to withstand high temperatures, and the high thermal inertia and high temperature stability inherent in the graphite core and support structures. These unique features have been exploited in the MHTGR to yield a reactor that does not depend on active, engineered safety features or human actions for safety of the public or the investment.



Coated fuel particles remain intact at high temperatures.

Inert Coolant

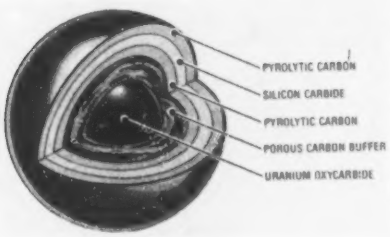
Helium gas, the reactor coolant, is inert so there are no chemical reactions between the coolant and the fuel, graphite core structure, or reactor vessel which could lead to fires or explosions. Further, helium has no effect on the nuclear reaction.

Stable, High Heat Capacity Core

The graphite fuel elements and reactor internals which make up the massive reactor core have a high heat capacity and maintain strength to temperatures beyond 5000°F. As a result, temperature changes of the core occur slowly and without damage to the core structure even in the event of accidents where the capability to remove heat is impaired.

Refractory Coated Fuel

In use for over 20 years, the refractory coated fuel has demonstrated the capability of retaining fission products under even severe conditions. Each fuel particle consists of a tiny uranium oxycarbide fuel kernel (about 350 microns in diameter) first coated with a porous graphite buffer, followed by three successive layers of pyrolytic carbon, silicon carbide and pyrolytic carbon. The three layer coating retains the fission products within the fuel kernel and buffer region. The outer diameter of the coated fuel particle is about 800 microns. Similar but slightly larger coated particles containing a kernel of thorium oxide are also used in the reactor. The coated particles are mixed with a graphitic material, formed into fuel rods, and then inserted in holes drilled in the graphite fuel elements blocks.



Refractory coatings encapsulate fuel kernel.

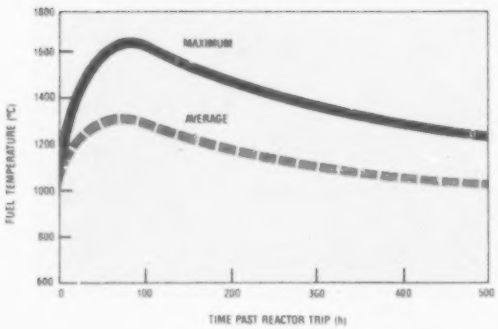
With the fuel rods in relatively close contact with the graphite blocks at all times, the massive graphite moderator reflector and support structure thereby provide an immediately available large heat sink during transient or emergency conditions.

Extensive operation and test data on coated particle fuel confirm that essentially no failure of the refractory coatings occurs if the fuel is maintained below 1800°C (3272°F).

Passive Cooling of Core

The annular core geometry, core power density, and the total module power of the MHTGR have been chosen such that the decay heat generated within the core can be passively removed by means of conduction, radiation, and natural convection without the fuel reaching a temperature at which the refractory coating would fail during an accident.

Primary cooling systems and shutdown cooling systems are available to remove heat and the core temperatures do not exceed normal levels. Even if both active cooling systems are unavailable, decay heat is dissipated by conduction and radiation to the reactor cavity cooling system (RCCS) in the reactor enclosure. The RCCS is a passive system that employs natural convection to transport the heat to an external sink. The heat transfer is sufficient to avoid damage to the fuel or reactor components. The most significant point, however, is that even if the RCCS is unavailable, passive radiation and conduction of heat to the silo structure and surrounding earth will limit maximum fuel temperatures to about 1600°C (2912°F), well below the failure temperature.



Accidents Mitigated by Inherent Features

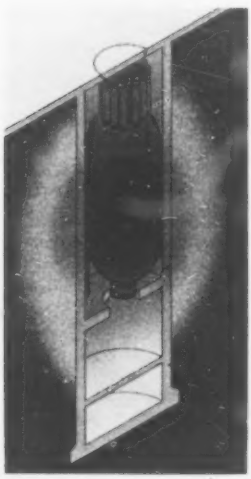
The inherent characteristics of the MHTGR also assure the safety of the system against other types of accidents.

Anticipated Transients Without Scram (ATWS) are mitigated by the strong negative temperature coefficient and zero coolant void coefficient of the reactor core. These inherent characteristics cause the reactor to automatically shutdown in the event reactor temperature rises above normal as a result of equipment failure or operator error.

Chemical reactions between steam and graphite can only occur as a result of accidents or equipment failures allowing the steam or water to come into contact with the graphite. At normal operating temperatures the reaction rate is insignificant. The reaction rate increases with temperature but, in any event, is highly endothermic (heat absorbing) and therefore self terminating.

Air/oxygen reactions with graphite are exothermic (heat releasing) but can only be sustained if multiple breaches of the reactor vessel occur and if the graphite is heated to temperatures much higher than normal. The silo installation limits the air available to support combustion in the extent of any reaction would be minor. Even were the graphite to be burned away the fission product activity is retained by the refractory particle coatings.

No public evacuation or sheltering is required even for severe low probability accidents because the consequences are mitigated by the inherent passive features of the MHTGR.



Below ground silo installation allows radiation and conduction of afterheat to the earth and restricts air access to the reactor.

Economics – Competitive for the Market

Passive Safety is Key To Modular Economics

Modular HTGR Economics Benefit From:

- Elimination of Excess Engineered Safety Systems
- Nonsafety Classification of Balance of Plant
- Plant Demand, Standardized Design
- Shortened Schedules
- Simplified Factory Fabrication
- Multiple Units on a Site

The inherent safety characteristics of the MHTGR provide the key to the economic potential of this small nuclear power plant. Many of the systems required for current types of reactors to provide adequate safety or investment protection can be eliminated. Further, since the safety of the MHTGR is independent of any system outside of the nuclear island, separated construction of the balance of plant and the use of conventional rather than nuclear standards for the majority of field construction should be permitted. The size of the nuclear equipment, including the steel vessels, is amenable to serialized factory fabrication and assembly and improved assurance of quality control and adequate pre operational testing. These features will lead to improvements in productivity and reduce the plant construction schedule and field erection costs.

The MHTGR is ideally suited for standardization of the nuclear design and to a design certification process for all equipment within the confined safety boundary. The licensing review for standard, replicate plants would be limited in scope and performed expeditiously, allowing a significant reduction in overall schedules.

Forecasts of the economics of commercial MHTGRs indicate overnight capital costs for a 4 module plant of less than $2000/kWe. The busbar cost of electricity appears to be over 10% less than the cost from an equivalent sized modern coal fired plant. These estimates indicate the MHTGR as a viable economic alternative to satisfy the future electric generation needs.

Domestic and International Incentives

The Modular High Temperature Gas Cooled Reactor offers a number of incentives which assure it a significant role in meeting future energy needs both domestically and internationally.

The MHTGR addresses the key domestic issues that must be overcome if nuclear energy is to assume a major energy role when current over capacity and load growth constraints have passed. It can provide economical energy costs, obtainable project costs and schedules, shortened and flexible planning cycles, reduced financial requirements and inherent and passive safety of the public and the investment.

The MHTGR is ideally suited for the international market, particularly for those developing nations with continuing and rising demand for electrification. The smaller, simpler easier to operate and maintain MHTGR reduces the requirement to develop a large, expensive support infrastructure thus permitting a more rapid deployment and beneficial return from nuclear energy.

Meets User Needs

The challenge for the nuclear industry is to develop options for potential customers that will enable them to make prudent choices for economic and competitive power and to provide a diverse, secure supply system for the future.

The MHTGR appears to be able to clearly meet this challenge for both domestic and international markets. The MHTGR meets or exceeds the established user requirements, has flexibility in size and end use to satisfy multiple market needs, is simple to operate and forgiving of mistakes, and most importantly, inherently and passively protects the health of public and the owner's investment.

HTGR Development Activities Worldwide

The High Temperature Gas Cooled Reactor (HTGR) is an advanced, second generation nuclear power system which employs several unique features. In particular, the HTGR core is cooled with pressurized helium, moderated and reflected with graphite, and fueled with tiny, refractory coated fuel particles.

Fort St. Vrain – U.S. Demonstration HTGR

United States

GA Technologies, then the General Atomic Division of General Dynamics Corporation, was the industrial pioneer of the gas-cooled concept in the U.S. dating back to 1957.

The initial HTGR project in the U.S. was the 40 MWe Peach Bottom I developmental plant that was placed on the system of the Philadelphia Electric Company. This plant was built as part of the U.S. Atomic Energy Commission's Power Reactor Demonstration Program and operated from June 1967 to October 1974 for 1349 equivalent full power days. It achieved an overall average nuclear steam supply system (NSSS) availability of 88%, excluding planned shutdowns for research and development programs. Peach Bottom I served as an invaluable test bed for HTGR fuel, core physics analyses, the hydrodynamics of a large helium system, fission product removal and operation of mechanisms in dry helium.

During the construction and startup of the Peach Bottom I reactor, design and development were proceeding on larger HTGRs. The Fort St. Vrain 330 MWe demonstration plant was initiated in 1965 as a follow-on part of the Power Reactor Demonstration Program. GA Technologies built the plant for Public Service Company of Colorado. The plant went critical in January 1974 and, as of the end of 1984, has produced over 3.9 billion kilowatt hours of net electrical energy. It also has provided important confirmation and demonstration of many inherent HTGR features. Fort St. Vrain contains a number of design features which are new to nuclear power systems in the U.S., namely, once-through modular steam generators with integral superheaters and reheaters, steam driven helium circulators, and a Prestressed Concrete Reactor Vessel (PCRV).

As for many first of a kind facilities, the achievement of commercial full power operation was delayed. In November 1981, the NRC granted permission to operate continuously at 100% power. Overall plant availability has suffered because of several instances of ingress of water from the helium circulator service system. However, most of the key components including the fuel, steam generators, control rod drives and the PCRV have performed well. Many of the design features of Fort St. Vrain are being directly incorporated in the current HTGR designs.

Summary Description
Fort St. Vrain HTGR

Thermal Power	842 MW(t)
Electrical Output	330 MW(e)
Design Net Efficiency	39.2%
Steam Conditions	2400 psi/1000°F
Core Exit Helium Temp.	1445°F
Average Fuel Temp.	1830°F
Maximum Fuel Temp.	2370°F
Power Density	6.3 Watts/cc
Average Burnup	100,000 MWd/tonne

United Kingdom

The gas cooled reactor concept originated in the United Kingdom. The Magnox, the first gas cooled reactor power station, began operation in 1956 at Calder Hall. It used carbon dioxide as a coolant and natural uranium as a fuel. There are now thirty four Magnox systems with a combined electrical rating of 7120 MW(e) in operation.

The Advanced Gas Cooled Reactor (AGR) was an evolutionary extension of the Magnox reactor. The system uses low-enriched uranium fuel in steel cladding that allows core outlet gas temperatures of 1200°F. In addition, the fuel element burn up rate is five times that of Magnox reactors and the core power density is three times greater. The AGR uses a graphite moderator, has integrated steam generators within a PCRV, and can be refueled under load. At present, six AGRs are operating, and eight are either in the construction or commissioning stage for a combined electrical power rating of 8840 MW(e).

The 20 MW(t) Dragon HTGR experimental reactor, a cooperative European project, was located in the U.K. The reactor operated successfully for 10 years commencing in 1964 and contributed substantially to the development of coated particle fuel used in all current HTGRs. Nearly 700 reactor years of U.K. operating experience form a key element in the gas-cooled reactor technology base.

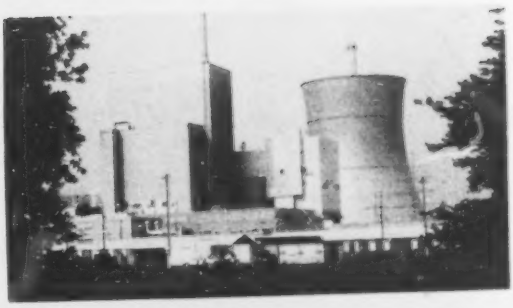
Germany

The HTGR program in the Federal Republic of Germany (FRG) builds on the very successful 15 MWe AVR developmental plant. The reactor has had an average availability of 68%. For extended periods since 1974, the core outlet temperature has been maintained at 1740°F.

The Germans are currently in the process of starting up the 300 MWe THTR demonstration plant. Criticality was achieved in

September 1983, and commercial operation is planned for late 1986. The AVR and THTR plants utilize a pebble bed core design in which the refractory coated fuel particles are disbursed in spherical graphite fuel elements that are continuously loaded and discharged through the core to effect on-line refueling.

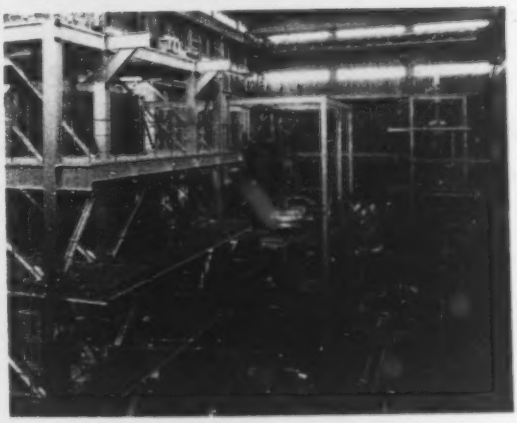
As a follow-on project candidate, a 500 MWe scale-up of the THTR, the HTR-500, is being advanced. In addition, there are ongoing efforts to develop a 200-250 MWt modular HTGR concept for small municipal utilities and cogeneration/process heat applications as well as for the export market. The German HTGR program is an important technical link for the U.S. that provides significant potential for international cooperation, thus reducing the overall development cost for the HTGR.



THTR – Federal Republic of Germany Demonstration HTGR

Japan

The Japanese have been pursuing the development and conceptual design of a very high temperature gas-cooled reactor (VHTR). The aim of the VHTR program is to demonstrate utilization of high temperature nuclear heat for process applications. The design of experimental VHTR (1740°F) test plants are currently being evaluated. Several major experimental facilities have been built and are operating, including the helium engineering demonstration loop (HENDEL), for testing high temperature components such as an intermediate heat exchanger. The operation of the experimental VHTR is scheduled to begin in the mid 1990s time frame, and the timing for the introduction of a commercial VHTR is around the year 2010.



HENDEL – Japanese helium loop used for testing large equipment in high temperature helium

Union of Soviet Socialist Republics

The USSR also has an ongoing program for the HTGR as well as the gas-cooled fast breeder reactor (GCFR). Several facilities have been built to study neutronics, thermal hydraulics, helium technology, and PCRVs. A design of an experimental pebble bed, steel vessel, 50 MWe HTGR, the VGR-50, has been completed. In addition to generating steam and electricity, the facility will permit various radiochemistry processes to be investigated. Progress has been made on the design of a 1000 MWt very high temperature (1740°F) pebble bed HTGR called the VG-400, housed in a multi-cavity PCRV. Besides producing electricity, the VG-400 will contain helium-to-helium intermediate heat exchangers for process heat and methane reforming applications.

User Involvement with HTGR

An underlying base of support throughout the history of the development of the High Temperature Gas Cooled Reactor has been a core of utility interest that has been organized for support of the various developmental and demonstration activities. This utility interest had its beginnings in the mid 1950's when a number of utilities and related industry formed the Rocky Mountain Nuclear Power Study Group to select a user preferred nuclear power concept. The HTGR was chosen based on the overall generation efficiency of the system, the capability to use modern fossil power conversion equipment, the inherent safety and low environmental impact characteristics and the reduced radiation exposure of plant personnel. Furthermore, the HTGR was seen as an evolutionary technology that extends to advanced applications for power generation and process heat.

Peach Bottom I – The First U.S. HTGR

In 1958, the Rocky Mountain Nuclear Power Study Group became the Rocky Mountain Pacific Nuclear Research Group to sponsor R&D on high temperature direct energy conversion, and on HTGR components and systems. Later in 1958, the High Temperature Reactor Development Associates (HTRDA), a group of 53 utilities, was formed to support the development and construction of the Philadelphia Electric Company system of the Peach Bottom I plant, the first HTGR built in the U.S. and part of the U.S. Atomic Energy Commission's (USAEC) Power Reactor Demonstration Program.

In 1960, the Empire State Atomic Development Associates, composed of seven New York State utilities, sponsored a three year design study for a 500 MWe HTGR and in 1964 and 1965 funded component development for larger HTGRs. In 1964, a group of 23 utilities sponsored a U/Th fuel and fuel cycle development program that established the basis for the fuel cycles used in the HTGR. The Advanced Reactor Development Associates Group, comprised of 11 utilities, was also formed in 1964 to sponsor conceptual studies on a fully integrated nuclear steam supply system within a prestressed concrete reactor vessel, which led to the Fort St. Vrain demonstration plant.

Fort St. Vrain – The U.S. Demonstration HTGR

In 1965, Public Service Company of Colorado signed contracts with GA Technologies (then the General Atomic Division of General Dynamics Corporation) and the USAEC for construction of a 330 MWe HTGR power plant at the Fort St. Vrain site near Denver, Colorado. Once construction of the Fort St. Vrain plant was well under way, GA Technologies commercially marketed 770 MW(e) and 1160 MW(e) HTGRs. Between 1971 and 1974, five twin unit plants were sold. During this period, considerable design detail and licensing experience materialized. In 1974 and 1975, these orders were cancelled or withdrawn because of the loss of market and commercial prospects prompted by the Arab oil embargo of 1973 and the resulting economic recession.

Gas-Cooled Reactor Associates (GCRA)

A number of studies and appraisals in the mid 1970's concluded that the HTGR technology was a desirable and viable future generation option, but that a new approach was needed to bring about the commercialization of the technology. Namely, greater involvement was needed by government and the end-user. The government embraced this concept and agreed to participate in the design and technology development, provided the end-users formally organized to assume a major role in guiding the development from the perspective of their end-user requirements.

Responding to this position, a group of six utilities officially formed Gas Cooled Reactor Associates in 1978. Since that time, the GCRA constituency has grown in spite of the overall setbacks for the nuclear industry and currently represents approximately one-third of the installed generation capacity in the U.S. In addition, the GCRA constituency includes several large industrial companies who are interested in cogeneration and process heat applications of the HTGR. GCRA utilities have a substantial investment in and experience with nuclear power, which provide a sound base for evaluating the HTGR as a commercial alternative.

The following is a current list of GCRA utilities (effective September 1986):

- American Public Power Assn
- Arizona Public Service Co.
- Baltimore Gas & Electric Co.
- Colorado Ute Electric Assoc.
- Delmarva Power & Light Co.
- Empire State Electric Energy Research Corp.
  - Central Hudson Gas & Electric
  - Consolidated Edison Co.
  - New York Power Authority
  - New York State Electric & Gas
  - Niagara Mohawk Power
  - Orange & Rockland Utilities
  - Rochester Gas & Electric Corp.
- Florida Power & Light Co.
- Gulf States Utilities
- Idaho Power Co.
- Northeast Utilities
  - Connecticut Light & Power
  - Holyoke Water Power Co.
  - Western Massachusetts Elec.
- Pacific Gas & Electric Co.
- Pacific Power & Light Co.
- Philadelphia Electric Co.
- Public Service Co. of Colorado
- Public Service Co. of New Mexico
- Public Service Electric & Gas Company
- Salt River Project
- San Diego Gas & Electric
- Tennessee Valley Authority
- Yankee Atomic Electric Co.

HISTORY OF UTILITY SUPPORT

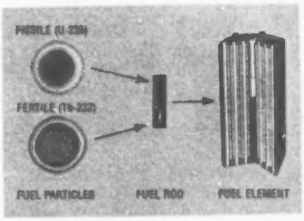
Name	Date	Major Contributions
Rocky Mountain Nuclear Study Group	1954	Evaluated potential nuclear alternatives. Chose the HTGR system for utility support
Rocky Mountain Pacific Nuclear Research Group	1958	Sponsored research and development on direct conversion and HTGR components and systems
High Temperature Reactor Development Associates	1958	Sponsored design and construction of 40 MWe Peach Bottom prototype HTGR
Empire State Atomic Development Associates	1960	Sponsored design of 500 MWe HTGR and significant research and development work on the major reactor components
Fuel and Fuel Cycle Development Program	1964	An extension of HTRDA, this group sponsored U/Th Fuel Cycle Development Program which established the basis for the fuel cycles used in the HTGR
Advanced Reactor Development Associates	1964	A natural outgrowth of HTRDA, this group sponsored conceptual studies on a fully integrated nuclear steam supply system within a pre-stressed concrete reactor vessel
Public Service Company of Colorado 330 MW(e) HTGR	1965	Public Service Company of Colorado signed contracts with General Atomic and USAEC to build the Fort St. Vrain Demo Plant
HTGR Gas Turbine Steering Committee	1973	Evaluated technical aspects of HTGR Gas Turbine Programs with respect to utility requirements
5 Commercial Orders for Steam Cycle Plants	1971/74	Detailed design and licensing experience, oil embargo of 1973 precipitated cancellations and commercial withdrawal by General Atomic
Helium Breeder Associates	1976	Supported development of Gas-Cooled Fast Breeder Reactor. Assets transferred to GCRA in 1982
Gas-Cooled Reactor Associates	1978	Supports development of gas-cooled reactors with current emphasis on modular HTGR. Provides utility/user requirements and overall coordination support for DOE's HTGR Program plus lead for demonstration project development activities

GCRA's Role

The primary function of GCRA is to represent Utility/User interests in the HTGR Program in establishing program objectives as well as the strategy to achieve those objectives. GCRA is responsible for guiding the development of the HTGR to be responsive to Utility/User requirements. This role ensures that the investments of government and private-sector resources for development will result in HTGR systems that will be valuable contributors to the U.S. energy economy. A full-time staff, supported solely by the utility funds, is in place to conduct Demonstration Project development activities, plus provide overall Program coordination support services to the U.S. Department of Energy (DOE). GCRA also interfaces with the Electric Power Research Institute (EPRI) to transfer technology from their HTGR Program and other EPRI-related programs to the HTGR Program. The goal of these activities is the development of HTGR systems that are optimally designed to meet the needs of the Utility User.

MHTGR Safety Features Have Been Demonstrated

Consideration of nuclear power plant safety focuses on those accidents in which the reactor core becomes overheated to the extent that barriers retaining the fission products are breached and radioactivity is released into the environment. The inherent and passive features of the MHTGR not only make such accidents less likely to happen, but also prevent the breaching of barriers and the release of radioactivity.



FUEL COMPONENTS

Core overheating is caused by the inability to:

- remove the heat generated in the nuclear fuel during normal operation, or
- remove the much lower level of decay heat that continues to be produced in the core even after shutdown, or
- limit the power produced by the reactor.

The keys to controlling overheating are to ensure that:

- the reactor can be shut down,
- power excursions are self-limiting,
- means are always available to remove the heat from the core,
- temperature increases are always slow and below any critical failure point, and
- there is a built-in margin that does not allow materials to react with each other at high temperatures.

These features have been demonstrated in the operating HTGR nuclear power systems and experiments conducted in both the United States and the Federal Republic of Germany.

Key Reactor Core Characteristics

The fuel for the HTGR consists of tiny, individually coated kernels of uranium oxycarbide or thorium oxide – stable ceramic materials. Each kernel is surrounded by a porous carbon layer and then encapsulated in three separate dense refractory layers consisting of an inner and outer layer of pyrocarbon sandwiching a layer of silicon carbide. The coated fuel particles which have been used successfully in the Fort St. Vrain AVR and THTR reactors for a number of years serve to contain and isolate the fission products produced during operation.

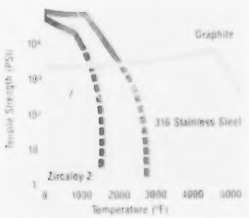
The coated particles are mixed with a graphite binder material to form finger-sized fuel compacts which are placed within hexagonal shaped graphite blocks to form fuel elements. Hundreds of fuel elements are stacked together and surrounded by similarly shaped graphite blocks which do not contain fuel to form the reactor core and reflector.

High Heat Capacity

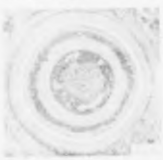
The graphite fuel elements and reactor internals which make up the massive reactor core have a high heat capacity. As a result, temperature changes of the core occur slowly even in the event of accidents where the capability to remove heat is lost or impaired, giving the operator a long time (measured in hours) to plan and conduct remedial action to prevent damage to the plant. The effectiveness of heat capacity in preventing rapid temperature buildup and damage to the reactor core or components (even loss of cooling) has been demonstrated at the AVR and Fort St. Vrain reactors.

High Temperature Stability

The ceramic-coated fuel particles and graphite fuel element and core structure have a unique high temperature resistance and can tolerate the temperatures that would be encountered in even severe accidents. Over twenty years of fuel development and fuel testing have confirmed that virtually no fuel failure and hence no fission product release occurs even if the coated particle fuel is heated up to temperatures of 3200°F. Unlike zirconium alloys and stainless steel used as fuel cladding or structural core materials in other reactors, graphite retains its structural strength up to temperatures of 4500°F.



GRAPHITE RETAINS STRENGTH AT HIGH TEMPERATURES



HTGR fuel particle after irradiation and heating at 1700°C. No damage to coating layers has occurred.

Inert Coolant

Helium gas, the reactor coolant, is inert so there are no chemical reactions between the coolant and the fuel, core structure or reactor vessel which could lead to fires or explosions. Further, helium has no effect on the nuclear reaction.

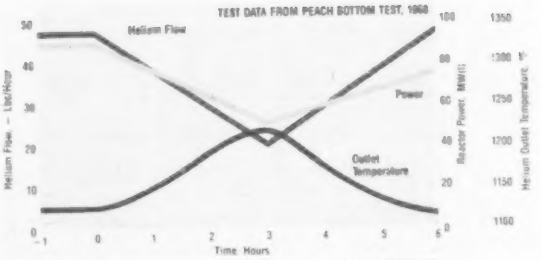
Tests Confirm Inherent Shutdown Characteristics

The neutronic characteristics of MHTGR assure that for all operating conditions if the core temperature increases above normal operating levels, the reactor heat generation rate will decrease, without the need for actuation of control systems. This desirable characteristic (negative temperature coefficient) has been demonstrated in various tests in the Peach Bottom and Fort St. Vrain reactors in the U.S. and in the AVR in Germany.

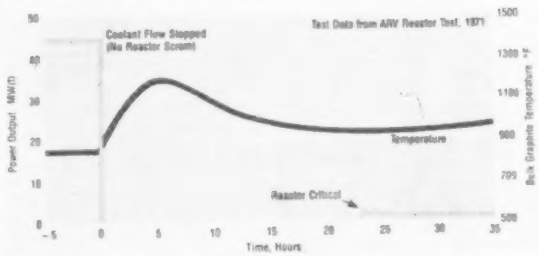
Reduced Flow Causes Power Decrease

In 1968 a series of tests were carried out at the Peach Bottom I HTGR to demonstrate the negative temperature coefficient. With the control rods fixed in position, the helium coolant flow was reduced to about one-half its original value, then increased again over a period of six hours. The inherent negative coefficient caused the reactor power to respond to the flow changes.

In 1978 a series of tests were run at the Fort St. Vrain HTGR reactor confirming the large negative temperature coefficient and its capacity to respond to transient conditions.



PEACH BOTTOM I TESTS CONFIRM REDUCING COOLANT FLOW REDUCES REACTOR POWER



NUCLEAR REACTION TERMINATES WITH LOSS OF COOLING

Loss of Cooling Causes Shutdown

The response to a severe accident condition was demonstrated with the AVR reactor by causing a loss of forced circulation condition. On September 22, 1970 at 6:05 AM, all circulators were shut down manually. All control rods were locked out of the core. Upon shutdown of the circulators, reactor power went to zero as a result of the negative temperature coefficient. On September 23, 1970 at 5:30 AM, i.e., 23.5 hours after start of the test, the reactor became critical again [as a result of the decay of xenon, a neutron absorbing material] reached a maximum power of only 1.8 Mw (4% of full power) and then decayed through a damped oscillation to an equilibrium power level of about 300 Kw (0.7% of full power). The test was terminated by insertion of the control rods two hours after achieving recriticality, i.e., 26 hours after start of the test. The test results clearly show that even if the control rods are not inserted following loss of circulation, there is no danger for the reactor. Even if no countermeasures are taken to prevent recriticality, the resulting power level is very low and does not lead to damage to the plant or the release of radioactivity from the fuel.

Small Reactor Size Limits Ultimate Core Temperature

The size and output of the MHTGR is important in limiting the ultimate temperature reached by the reactor core in an accident. In small reactors, relatively large areas are provided by the reactor vessel so that decay heat being generated in the reactor core can be removed by natural radiation, conduction, or convection, thus limiting the maximum core temperatures, even when normal cooling is lost. The output of the MHTGR has been chosen such that for any credible accident the integrity of the system is not threatened.

The MHTGR and Chernobyl

The Chernobyl nuclear disaster has raised questions regarding the design similarities and differences between the Chernobyl (RBMK type) Boiling Water Graphite Moderated Reactor, the Modular High Temperature Gas-Cooled Reactor (MHTGR), and other reactor systems. In particular, attention has been drawn to the use of graphite as a moderator.

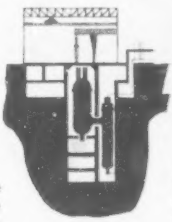
Heating of reactor grade graphite to 3000°F does not result in self-sustained burning.

Graphite has long been considered a means for improving reactor performance and safety because of its excellent nuclear and heat transfer properties, its stability at very high temperatures, and its capability to provide a large heat storage capacity. The MHTGR is designed to take advantage of these graphite properties to attain an inherent and passively safe system.

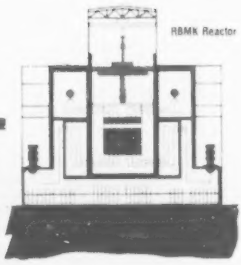
Official reports of the accident indicate that graphite was not the cause of the accident. Further study indicates that, even if a comparable sequence of operator errors occurred with the MHTGR, there would have been no explosion, no release of radioactivity from the system, and no need for evacuation of the plant or surrounding countryside.

Chernobyl Reactor

The Chernobyl RBMK reactor is a pressure tube type of reactor using zirconium alloy clad uranium oxide pellets as fuel, boiling light water as the coolant and graphite as the moderator. The fuel assemblies consist of a cluster of 18 clad fuel pins located in vertical zirconium alloy process tubes through which the cooling water is circulated. The fuel assemblies and coolant temperatures and pressures are similar to those for commerical boiling water reactors. The approximately 1700 process tubes which serve as the pressure boundary for the system pass through stacked graphite moderator blocks. The reactor is surrounded by concrete and steel shielding and support structures.



MHTGR Reactor



RBMK Reactor

Modular HTGR Reactor

The MHTGR uses refractory coated particle fuel, helium gas as a coolant, and graphite as the moderator and core structural material. The fuel particles consist of microspheres of uranium oxycarbide clad with pyrolytic carbon and silicon carbide which form a barrier and pressure vessel to retain fission products. These refractory coatings are stable to very high temperatures. The coated particles are bonded together with a graphitic binder and inserted into large graphite fuel element blocks. The helium gas passes through coolant holes in the fuel element blocks. The pressure boundary for the MHTGR system is a steel reactor vessel which is contained in a below ground silo.

Different Features Affect Safety

The significant differences between RBMK and HTGR reactors are reflected in their susceptibility and response to accidents.

Reactivity Accidents

The nuclear design of the Chernobyl core makes the reactor power level difficult to control under certain conditions. An accident leading to a loss of the water coolant and/or an increase in the graphite temperature can cause a very rapid increase in reactor power to steam formation and to overpressurization. In the MHTGR, the nuclear reaction is unaffected by the loss of helium coolant and increased fuel or graphite temperatures result in reduced power outputs and ultimately to the reactor shutting itself down.

The MHTGR has a negative temperature and power coefficients which protect the reactor against rapid power excursions.

Fuel Heatup Rate on Loss of Coolant

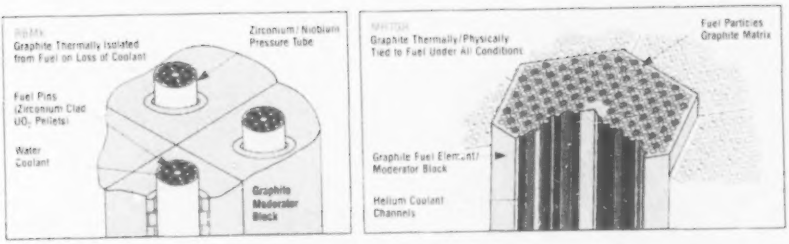
The heat up rate is slow for the MHTGR since it is determined by the heat capacity of the massive graphite moderator to which it is closely coupled.

The heat up rate is rapid for the RBMK since the heat capacity of the fuel assemblies is small and the fuel assemblies are thermally isolated from the graphite moderator.

Temperature Stability and Margins

The refractory MHTGR fuel particle coatings maintain their integrity and retain fission products to temperatures of over 3200°F. The graphite core structure gains in strength to 4500°F and is structurally sound to 5400°F.

The zirconium alloy RBMK fuel cladding loses strength and starts release of fission products at approximately 1300°F during a loss of coolant accident.


RBMK
Graphite Thermally Isolated from Fuel on Loss of Coolant
Zirconium / Niobium Pressure Tube
Fuel Pins (Zirconium Clad UO_2 Pellets)
Water Coolant
Graphite Moderator Block
MHTGR
Graphite Thermally/Physically Tied to Fuel Under All Conditions
Fuel Particles Graphite Matrix
Graphite Fuel Element/ Moderator Block
Helium Coolant Channels

Coolant Reactions

The MHTGR helium coolant is inert and does not react with graphite, fuel coatings, or other system components.

In the RBMK, an exothermic (heat producing) chemical reaction between the water coolant and the zirconium alloy fuel cladding is initiated at about 1800°F.

Graphite Burning

It is difficult to induce self-sustained burning of the large, dense graphite shapes used in either reactor. Under ideal controlled conditions, graphite temperatures in excess of 1300°F are required. Such conditions could result only as a consequence of a major accident which could both raise the temperature of the graphite moderator and cause multiple breaches of the barriers normally preventing air contact with the graphite. The consequences from such an accident are insignificant for the MHTGR.

The below grade level silo installation limits the quantity of air available to sustain combustion.

The silicon carbide fuel particle coating retains its capability to retain fission products, even if the graphite has burned away.

Wigner Energy Release

A buildup of radiation induced (Wigner) energy in graphite can lead to a rapid increase in temperature when released. This is not a consideration in the HTGR or RBMK reactor, since both reactors operate at temperatures at which the energy is annealed out of the graphite.

Water-Graphite Reactions

In either the RBMK or MHTGR reactor systems, water and graphite come in contact only as a result of equipment failures or accidents which breach a portion of the primary pressure boundary. At above normal operating temperatures, water and graphite reactions can lead to the formation of hydrogen and carbon monoxide, however the reaction is strongly endothermic (absorbs heat) and is self-limiting.

Loss of Cooling or Coolant Accidents

The nuclear accidents of major concern are those that can result in releases of fission products from the fuel and the failure of the pressure boundary, allowing the radioactive material to leave the system environment. These accidents would characteristically involve a loss of cooling or of the coolant.

In the RBMK, on the loss of coolant, if immediate corrective action is not taken, failure of the zirconium alloy clad and the release of fission products into the system can take place in a matter of minutes. Further damage to the cooling system and/or exothermic chemical reactions (such as the zirconium water reaction) can occur which will lead to a rupture of the system and to radiation release from the plant. For the MHTGR, the loss of cooling involves the loss of helium coolant pressure, along with the loss of all active means of circulating the coolant or of removing heat from the core. In the MHTGR the fuel temperature increases very slowly and sufficient heat can be removed from the core by passive radiation and conduction to the surroundings that temperatures never reach the limits at which the refractory fuel cladding will fail and release fission products within the system.

Inherent and Passive Safety

The passive safety inherent in the fuel system, coolant, and plant arrangement for the MHTGR, which are dramatically different from the RBMK type system, essentially preclude the possibility of an accident such as that reported at Chernobyl and, more importantly, prevent accident consequences which would result in the significant release of radioactivity from the plant or for the need of public evacuation.

COMPARED REACTOR FEATURES

TYPE REACTOR	RBMK	MODULAR HTGR
COOLANT	BOILING WATER	HELIUM GAS
MODERATOR	GRAPHITE BLOCKS	GRAPHITE BLOCKS
CONFIGURATION	VERTICAL PRESSURE TUBE	DUAL VESSEL
FUEL		
FUEL FORM	UO_2 PELLETS	UCO MICROSPHERES
FUEL CLADDING	ZIRCONIUM ALLOY	PYROCARBON/SILICON CARBIDE
CONFIGURATION	ASSEMBLIES	PARTICLE COMPACTS IN GRAPHITE BLOCKS
COOLANT		
MATERIAL	LIGHT WATER	HELIUM
PRESSURE, PSI	990	925
TEMPERATURE, IN/OUT °F	518/544	497/1058
PRESSURE BOUNDARY		
TYPE	PRESSURE TUBES/PIPING	REACTOR VESSEL/STEAM GEN
MATERIAL	ZIRCONIUM/ST. STL.	STEEL/ST. STL.

HTGR-86-095

DRAFT

HTGR

— SUMMARY — CONCEPTUAL DESIGN STATUS AND DEFINITION FOR THE MODULAR HTGR PLANT

AUTHORS/CONTRACTORS

BECHTEL NATIONAL, INC.
COMBUSTION ENGINEERING
EG&G IDAHO, INC.
GA TECHNOLOGIES INC.
GAS-COOLED REACTOR ASSOCIATES
GENERAL ELECTRIC CO.
OAK RIDGE NATIONAL LABORATORIES
STONE & WEBSTER

ISSUED BY GA TECHNOLOGIES INC.
FOR THE DEPARTMENT OF ENERGY
CONTRACT DE-AC03-84SF11963

SEPTEMBER 1986

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HTGR–86–095
DRAFT

— SUMMARY —

CONCEPTUAL DESIGN STATUS AND DEFINITION FOR THE MODULAR HTGR PLANT

Issued By:
GA Technologies Inc.
P.O. Box 85608
San Diego, California 92138

DOE Contract No. DE–AC03–84SF11963

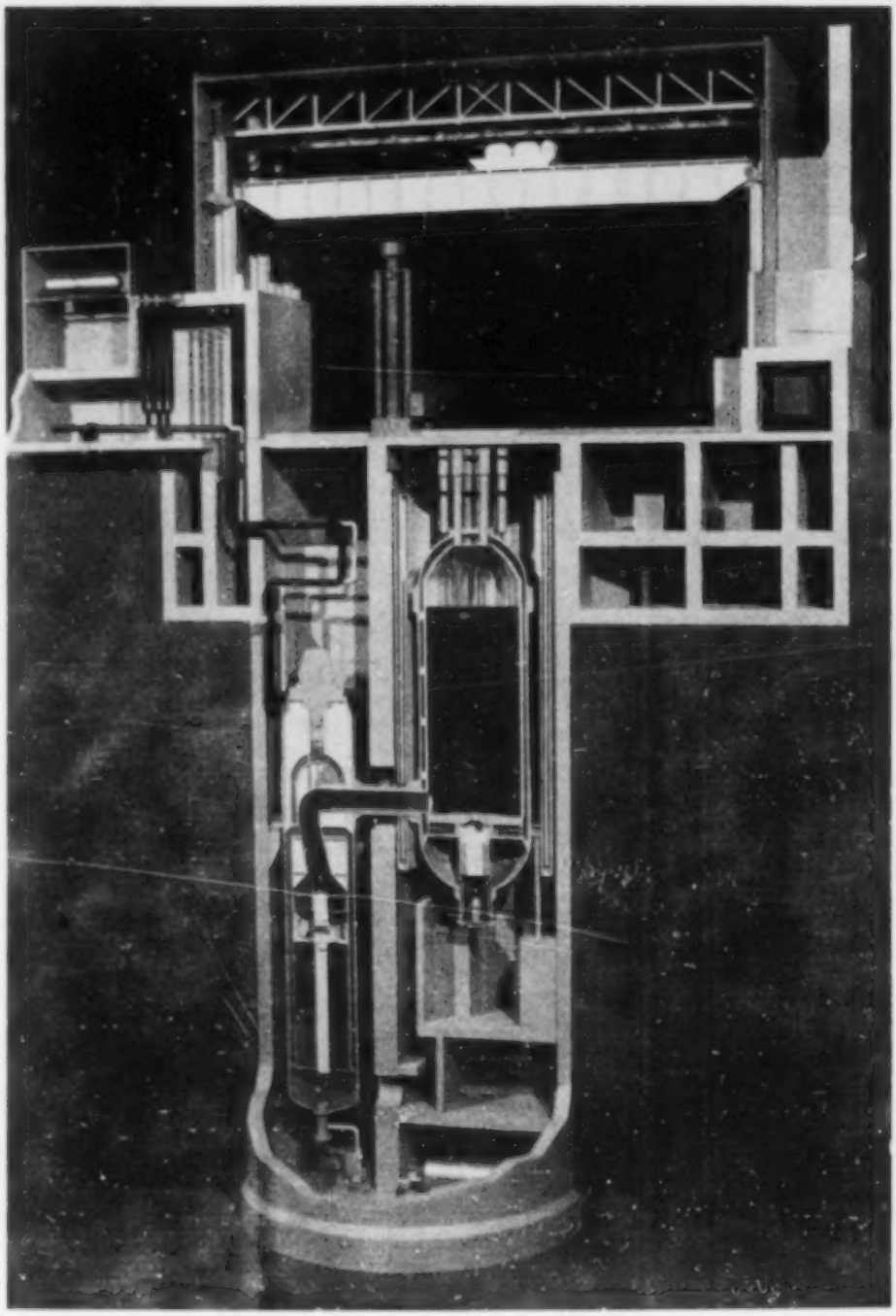
SEPTEMBER 1986

SUMMARY - CONCEPT DESIGN STATUS AND DEFINITION

- CONTENT -

- Meeting the needs of the nuclear industry
- Major accomplishments in FY'86
- Design Definition
  - Plant
  - NSS (by major systems/components)
- Technology (fuel, material)
- Licensing interactions
- Safety assessments
- Fuel cycle requirements
- Economics/cost estimate
- Schedule
- Market interest/private sector support
- International cooperation
- Advanced applications of MHTGR
- Future milestones

Appendix A - Conceptual Design Data



MEETING THE NEEDS OF THE NUCLEAR INDUSTRY

MODULAR HTGR DOES ADDRESS THE INDUSTRY PROBLEMS AND CHANGING NEEDS

- PUBLIC AND INSTITUTIONAL ACCEPTANCE
  - INHERENT SAFETY
  - LOW INVESTMENT RISK
- REDUCED COST/SCHEDULE UNCERTAINTY
- REDUCED FINANCIAL COMMITMENTS
- SMALLER POWER ADDITIONS
- SHORTER PLANNING CYCLES
- COMPETITIVE ECONOMICS

THE MODULAR HTGR

- STANDARDIZED, PRE-LICENSED, FACTORY FABRICATED REACTORS
- STANDARD HTGR FEATURES
  - HELIUM GAS COOLANT
  - GRAPHITE MODERATOR/CORE STRUCTURE
  - CERAMIC CLAD FUEL
  - HIGH TEMPERATURE/HIGH EFFICIENCY
  - LOW RADIOLOGICAL WASTES, LOW WORKER EXPOSURE
- SPECIAL MODULAR FEATURES
  - DEMONSTRABLE PASSIVE SAFETY
  - SMALL UNIT RATING (INCREMENTAL POWER ADDITIONS)

MODULAR HTGR
MEETS UTILITY USER REQUIREMENTS

- PLANT SIZE ~500 MW(e)
- EQUIVALENT AVAILABILITY >80%
- PROBABILITY OF REACTOR MODULE LOSS $<10^{-5}$/MODULE-YR
- MEET EXISTING SAFETY AND LICENSING CRITERIA WITH NO PUBLIC SHELTERING FOR EVENT FREQUENCIES $\sim 5 \times 10^{-7}$/YEAR
- 10% POWER COST ADVANTAGE OVER COAL

THE MODULAR HTGR ASSURES PUBLIC SAFETY

- WITHOUT OPERATOR ACTIONS
- WITHOUT OPERATION OF ANY POWERED SAFETY SYSTEMS
- WITHOUT THE NEED OF PUBLIC SHELTERING

NEAR TERM PROGRAM OBJECTIVE IS A DEMONSTRATION PROJECT

- FULL SCALE, SINGLE MODULE POWER PLANT
- DEMONSTRATE PERFORMANCE, SCHEDULE, ECONOMICS
- DESIGN CERTIFICATION
- 10 YEAR, $800 MILLION PROJECT

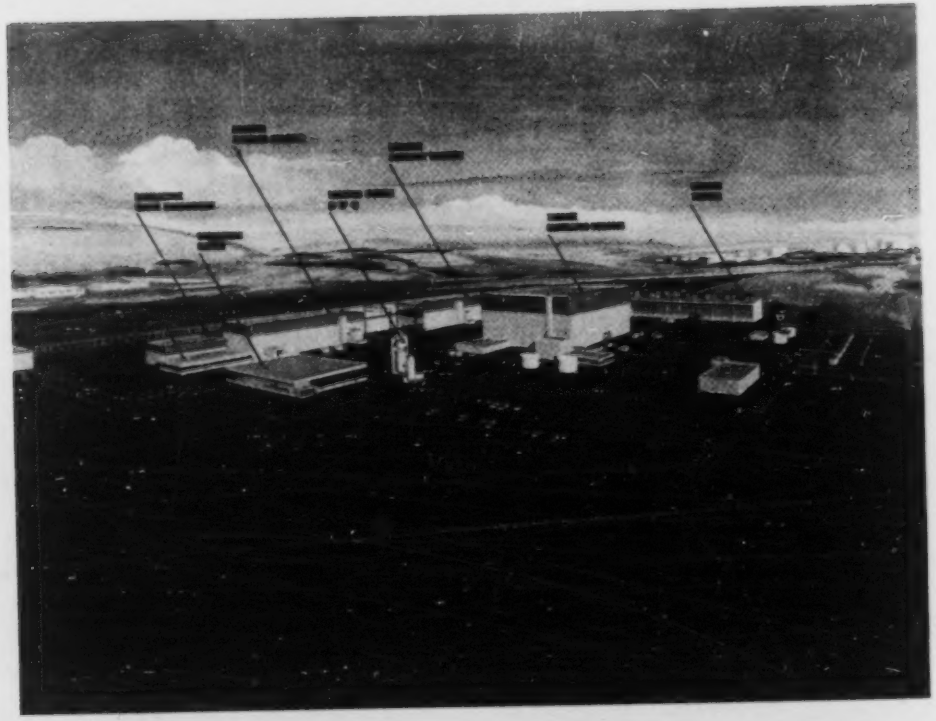
MAJOR ACCOMPLISHMENTS IN FY '86

FY-85/FY-86 MHTGR PROGRAM ACCOMPLISHMENTS

- TOP LEVEL USER REQUIREMENTS AND PROGRAM GOALS ESTABLISHED
- ALTERNATIVE DESIGN EVALUATION COMPLETED
  - 350 MW(t) MODULAR HTGR WITH ANNULAR PRISMATIC CORE SELECTED
  - ECONOMIC, INVESTMENT PROTECTION AND SAFETY AND LICENSING CRITERIA CAN BE MET
- LICENSING PLAN ESTABLISHED WITH NRC
- TOP LEVEL REGULATORY CRITERIA SUBMITTED TO NRC, REVIEWS COMPLETED
- PRELIMINARY SAFETY INFORMATION DOCUMENT (PSID) TO BE SUBMITTED 9/86
- PROJECT DEVELOPMENT INITIATIVE UNDERTAKEN BY GCRA AND OTHER INDUSTRIAL PARTICIPANTS

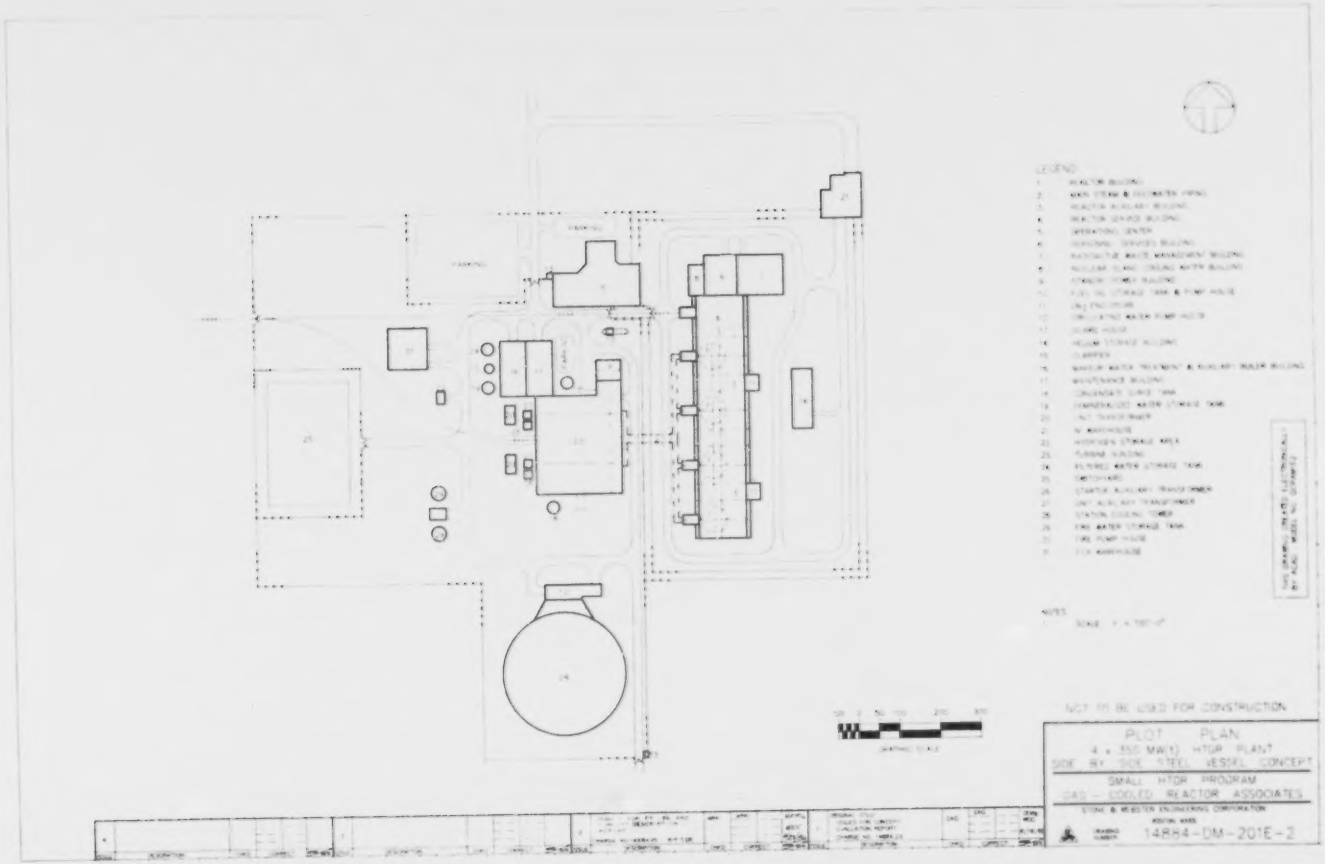
DESIGN DEFINITION

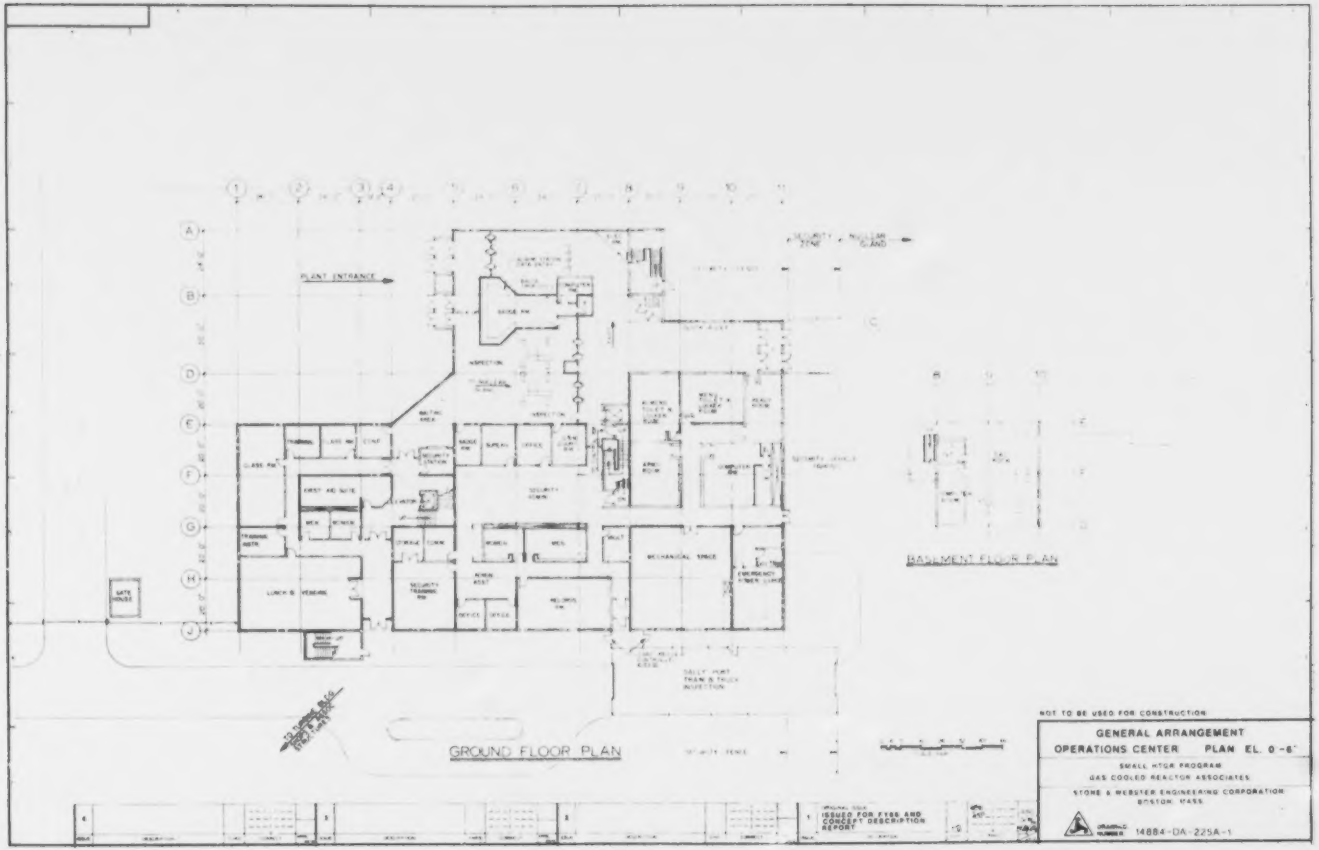
FOUR-MODULE MHTGR POWER PLANT

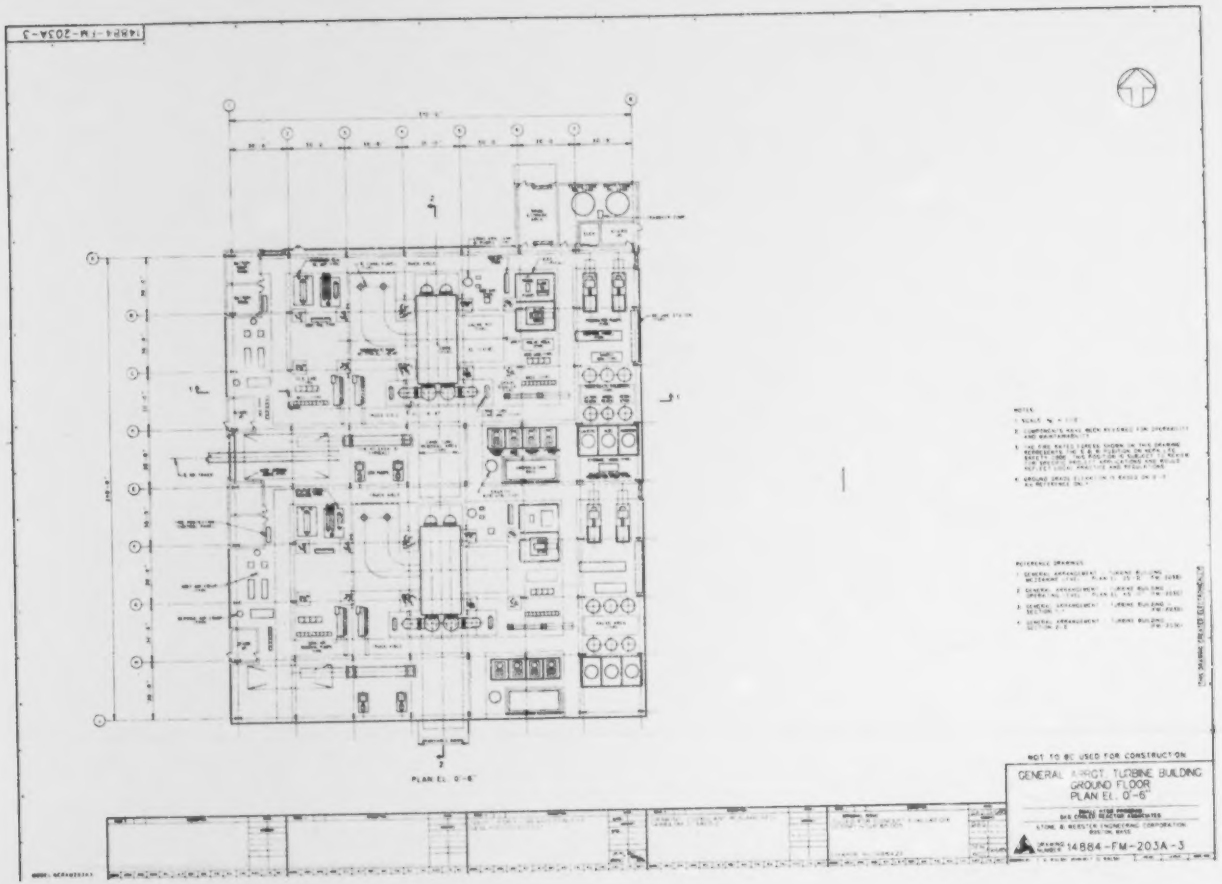


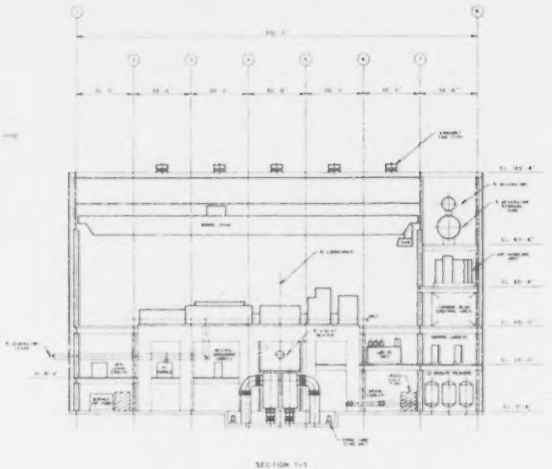
THE MODULAR HTGR HAS HIGH THERMAL EFFICIENCY, LOW URANIUM NEEDS

NUCLEAR SYSTEM PARAMETERS	
THERMAL POWER, MW(t)	350
HELIUM PRESSURE, PSIA	925
HELIUM TEMPERATURE, °F	490/1270
POWER SYSTEM PARAMETERS	
POWER CYCLE	NON-REHEAT
TURBINE INLET PRESSURE/ TEMPERATURE, PSIA/°F	2400/1000
SYSTEM PARAMETERS	
NET ELECTRICAL OUTPUT, MW(e)	135
NET THERMAL EFFICIENCY, %	39


NOT TO BE USED FOR CONSTRUCTION
PLOT PLAN
4 x 350 MW(t) HTGR PLANT
SIDE BY SIDE STEEL VESSEL CONCEPT
SMALL HTGR PROGRAM
GAS - COOLED REACTOR ASSOCIATES
STONE & WEBSTER ENGINEERING CORPORATION
14R84-DM-201E-2


PLANT ENTRANCE
GATE HOUSE
GROUND FLOOR PLAN
BASEMENT FLOOR PLAN
NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENT
OPERATIONS CENTER PLAN EL. 0'-6"
SMALL HTGR PROGRAM
GAS COOLED REACTOR ASSOCIATES
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
14884-DA-225A-1
ISSUED FOR FDSS AND CONCEPT DESCRIPTION REPORT


14884-FM-203A-3
PLAN EL. 0'-6"
NOTES
REFERENCE DRAWINGS
NOT TO BE USED FOR CONSTRUCTION
GENERAL
PROT. TURBINE BUILDING
GROUND FLOOR
PLAN EL. 0'-6"
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
14884-FM-203A-3

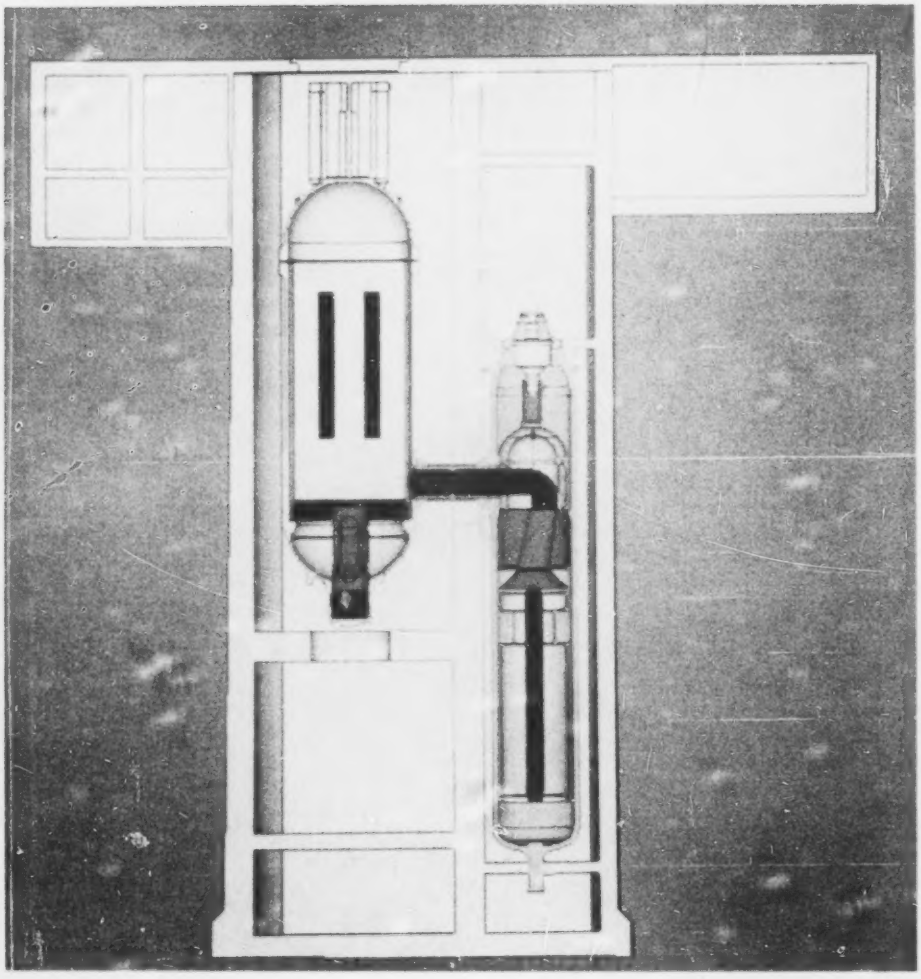


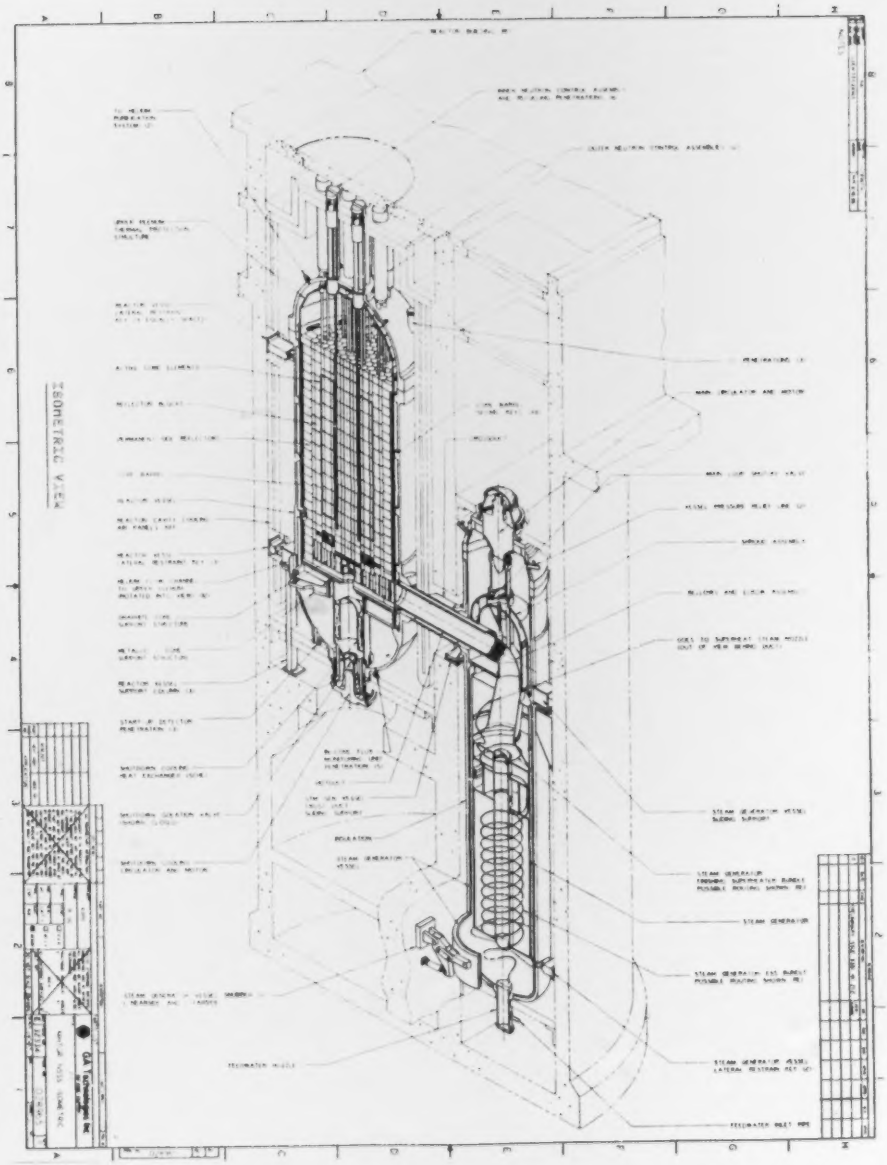
NOT TO BE USED FOR CONSTRUCTION

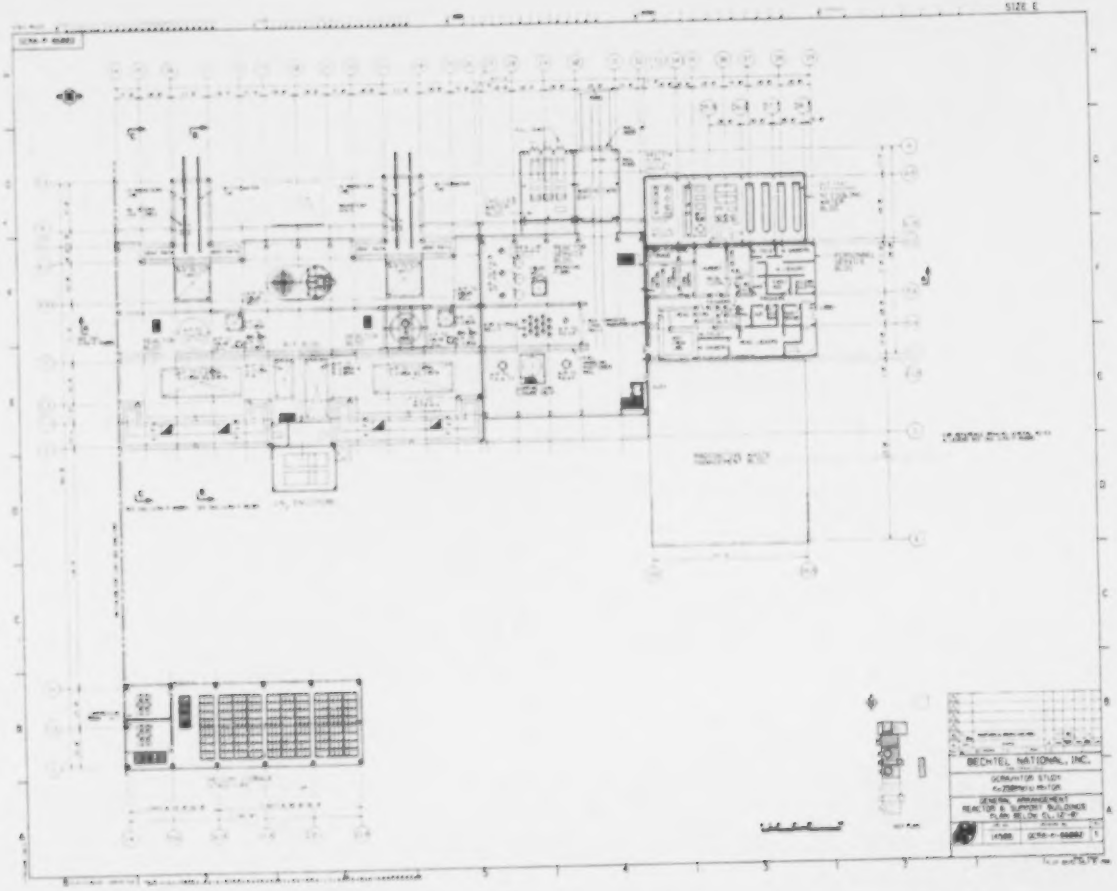
GENERAL ARRGT TURBINE BUILDING
SECTION 1-1

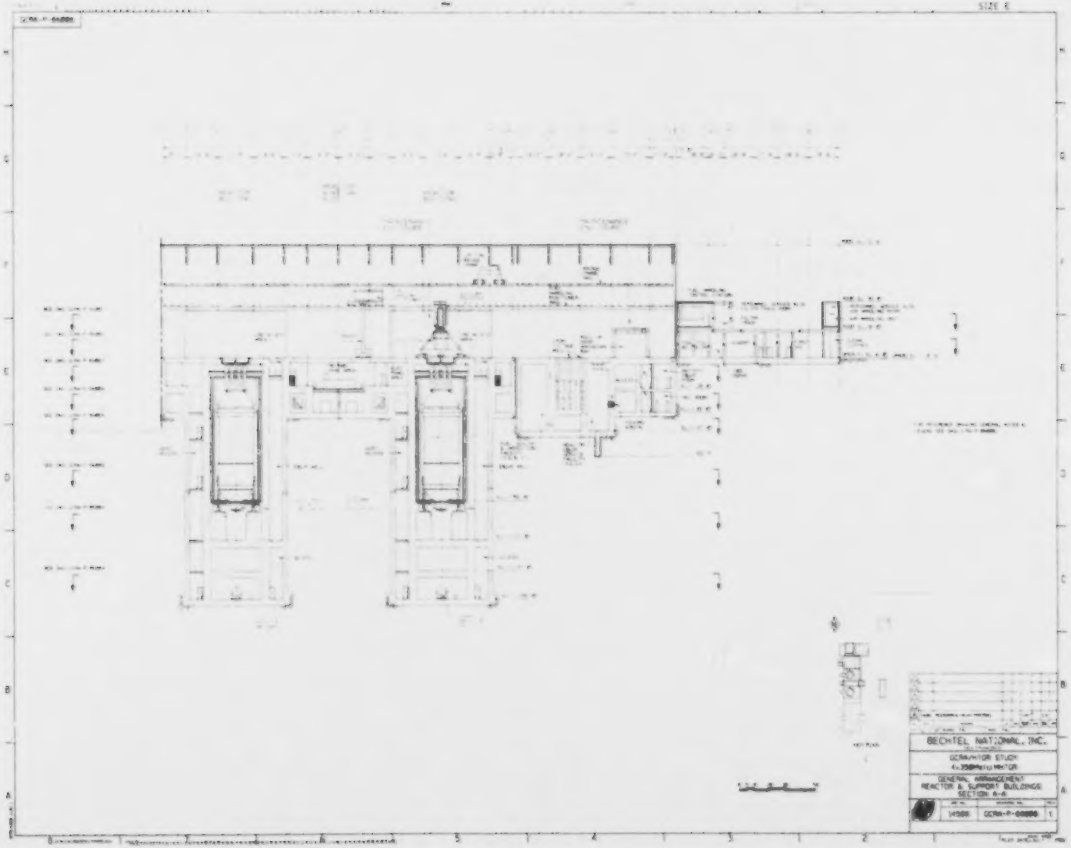
DRAWING NUMBER 14884-FM-2030-3

MODULAR HTGR IN SILO




ISOMETRIC VIEW


SIZE E
BECHTEL NATIONAL, INC.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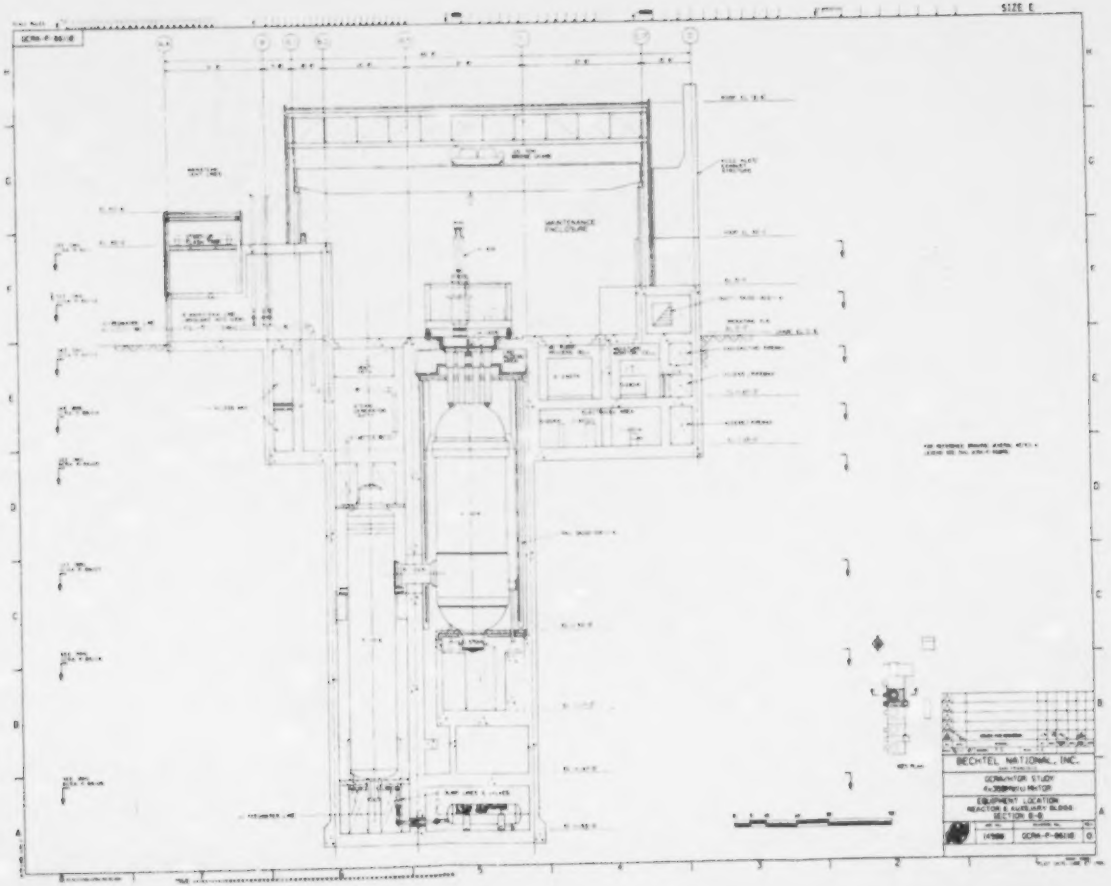
BECHTEL NATIONAL, INC.
GENERAL ARRANGEMENT
REACTOR & SUPPORT BUILDINGS
SECTION A-A


SIZE E
MAINTENANCE
BECHTEL NATIONAL, INC.
EQUIPMENT LOCATION
REACTOR & AUXILIARY BLDGS.
SECTION B-B
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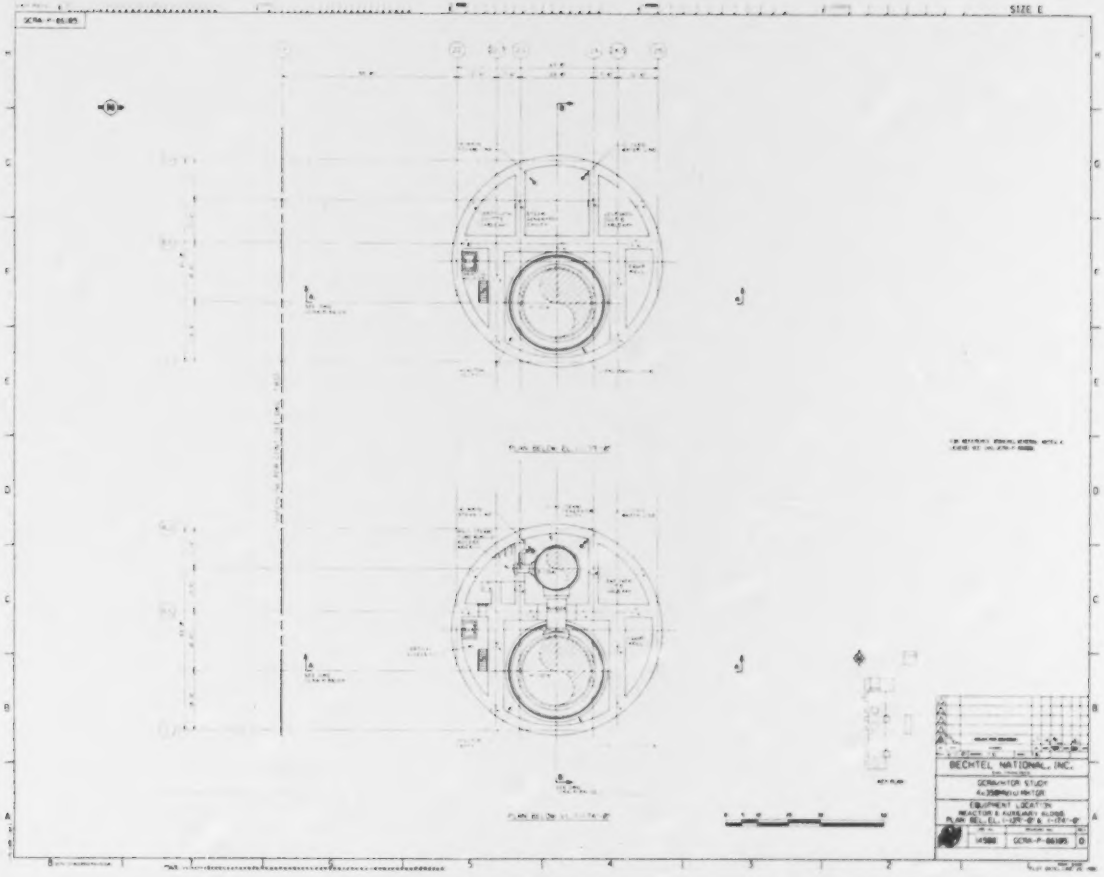
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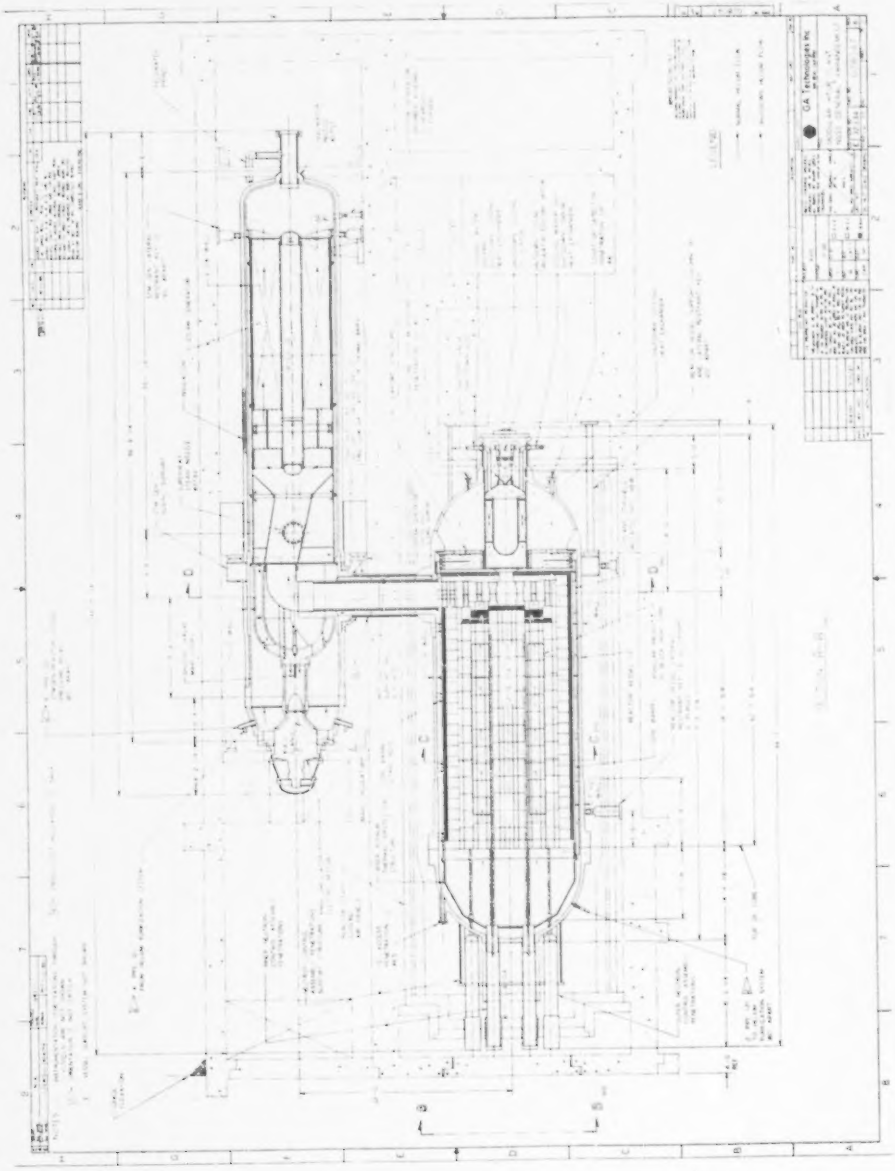
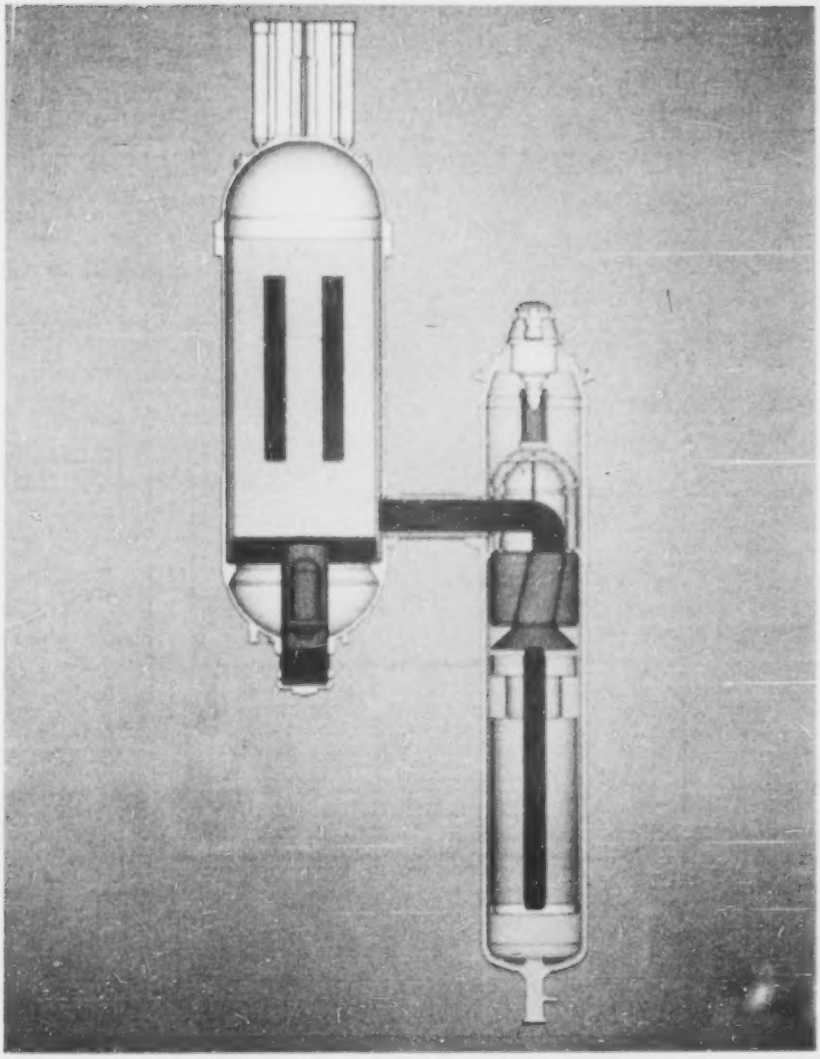
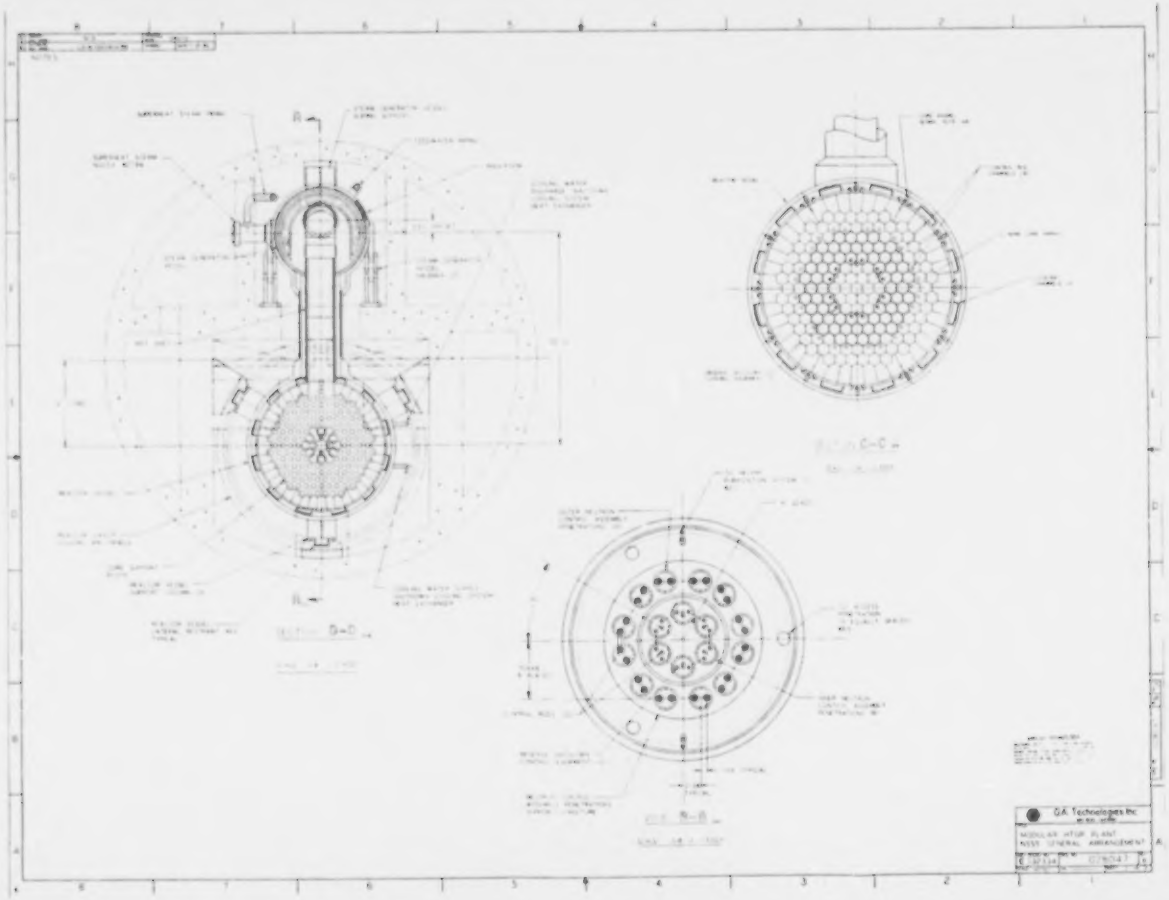
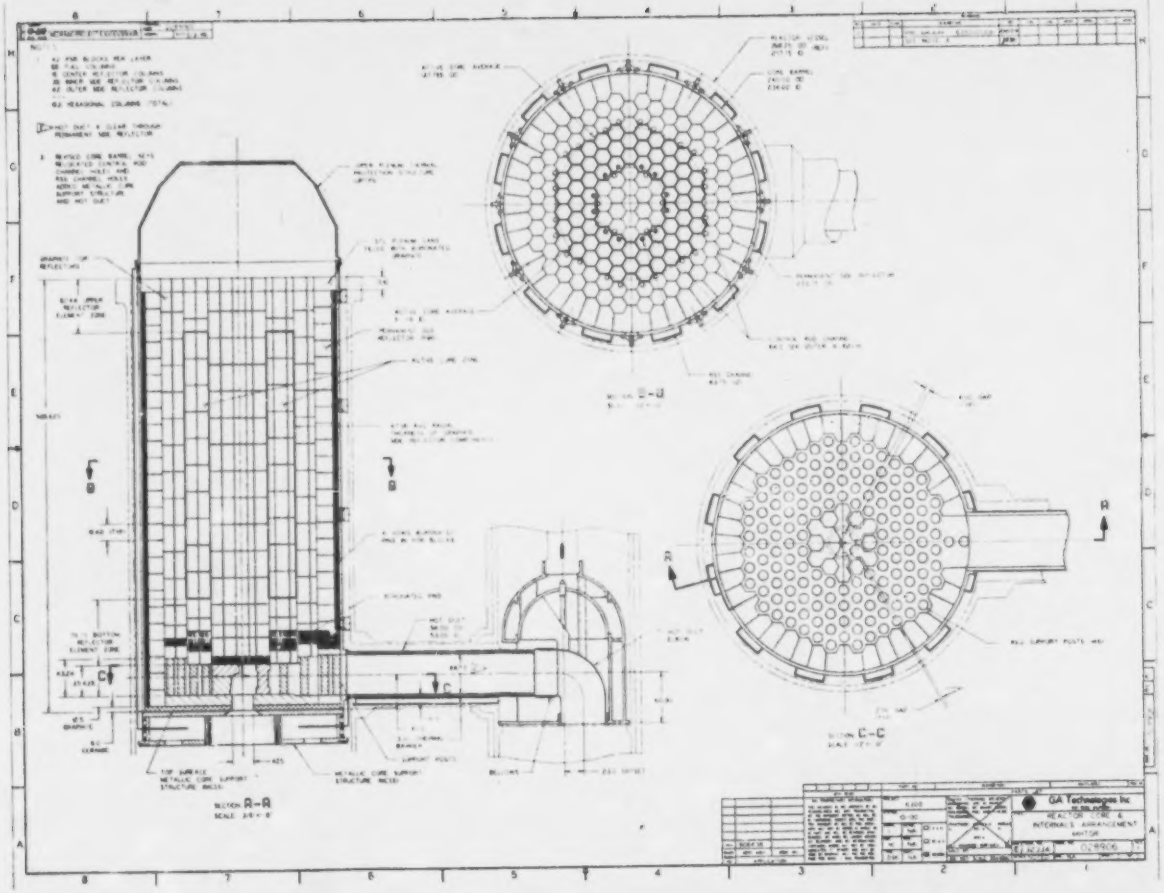
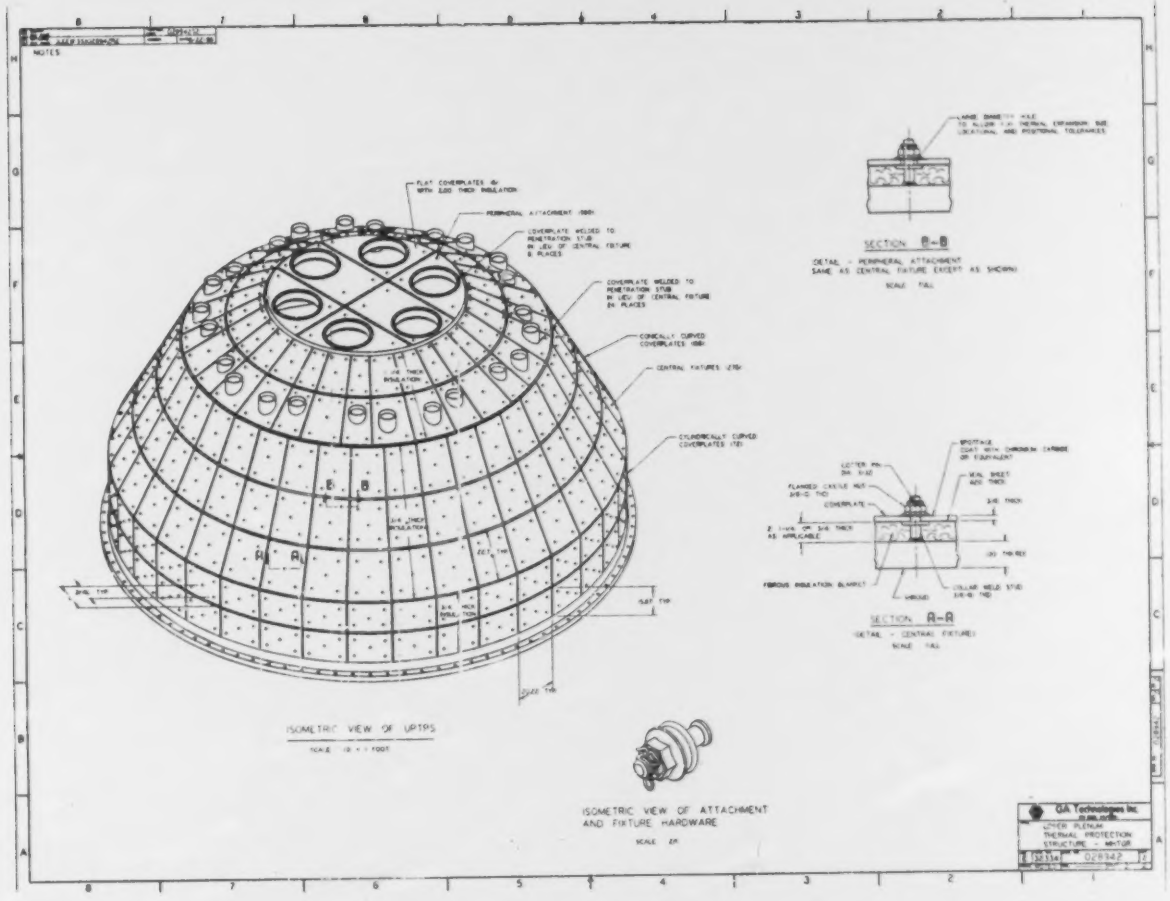


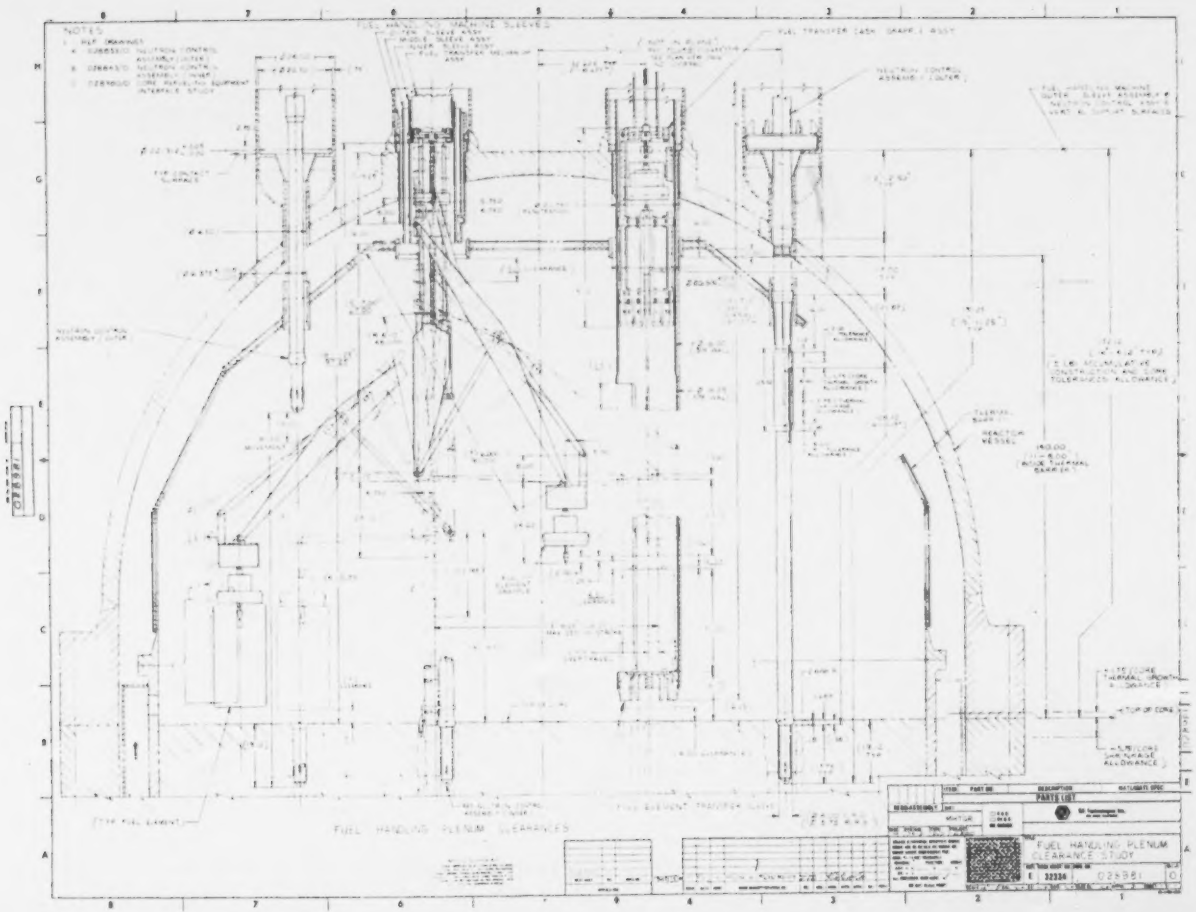
MODULAR HTGR WITH ANNULAR PRISMATIC CORE

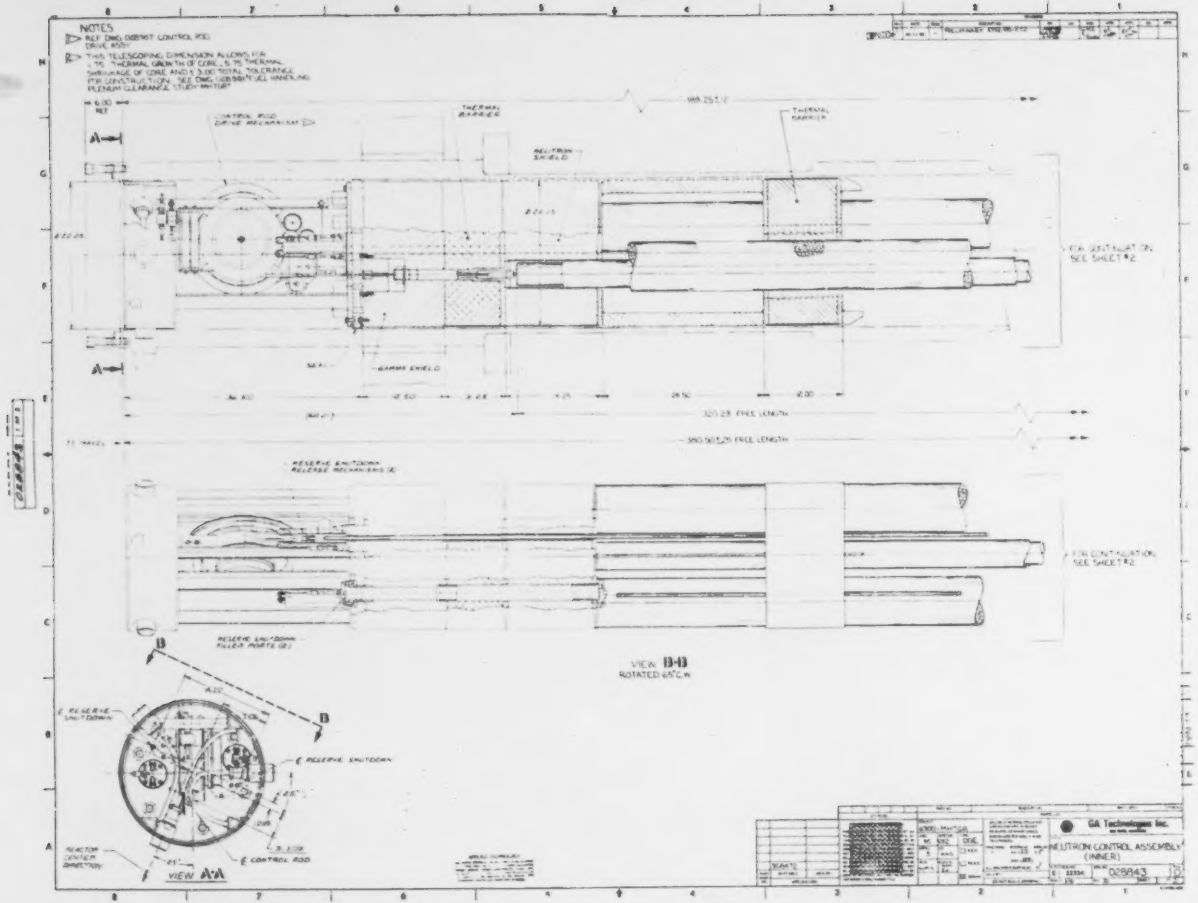



GA Technologies Inc.


SECTION A-A
SECTION B-B
SECTION C-C
GA Technologies Inc.


NOTES
FLAT COVERPLATES W/ WITH 3.00 THICK INSULATION
PERIPHERAL ATTACHMENT (TYP)
COVERPLATE WELDED TO PENETRATION STUB IN LIEU OF CENTRAL FIXTURE 8 PLACES
COVERPLATE WELDED TO PENETRATION STUB IN LIEU OF CENTRAL FIXTURE 24 PLACES
CONICALLY CURVED COVERPLATES (TYP)
CENTRAL FIXTURES (TYP)
CYLINDRICALLY CURVED COVERPLATES (TYP)
SECTION B-B
DETAIL - PERIPHERAL ATTACHMENT SAME AS CENTRAL FIXTURE EXCEPT AS SHOWN
SCALE FULL
SECTION A-A
DETAIL - CENTRAL FIXTURE
SCALE FULL
ISOMETRIC VIEW OF UPTPS
SCALE
ISOMETRIC VIEW OF ATTACHMENT AND FIXTURE HARDWARE
SCALE
GA Technologies Inc.
LOWER PLENUM
THERMAL PROTECTION
STRUCTURE - MINTOR


FUEL HANDLING PLENUM CLEARANCES
PARTS LIST
FUEL HANDLING PLENUM CLEARANCE STUDY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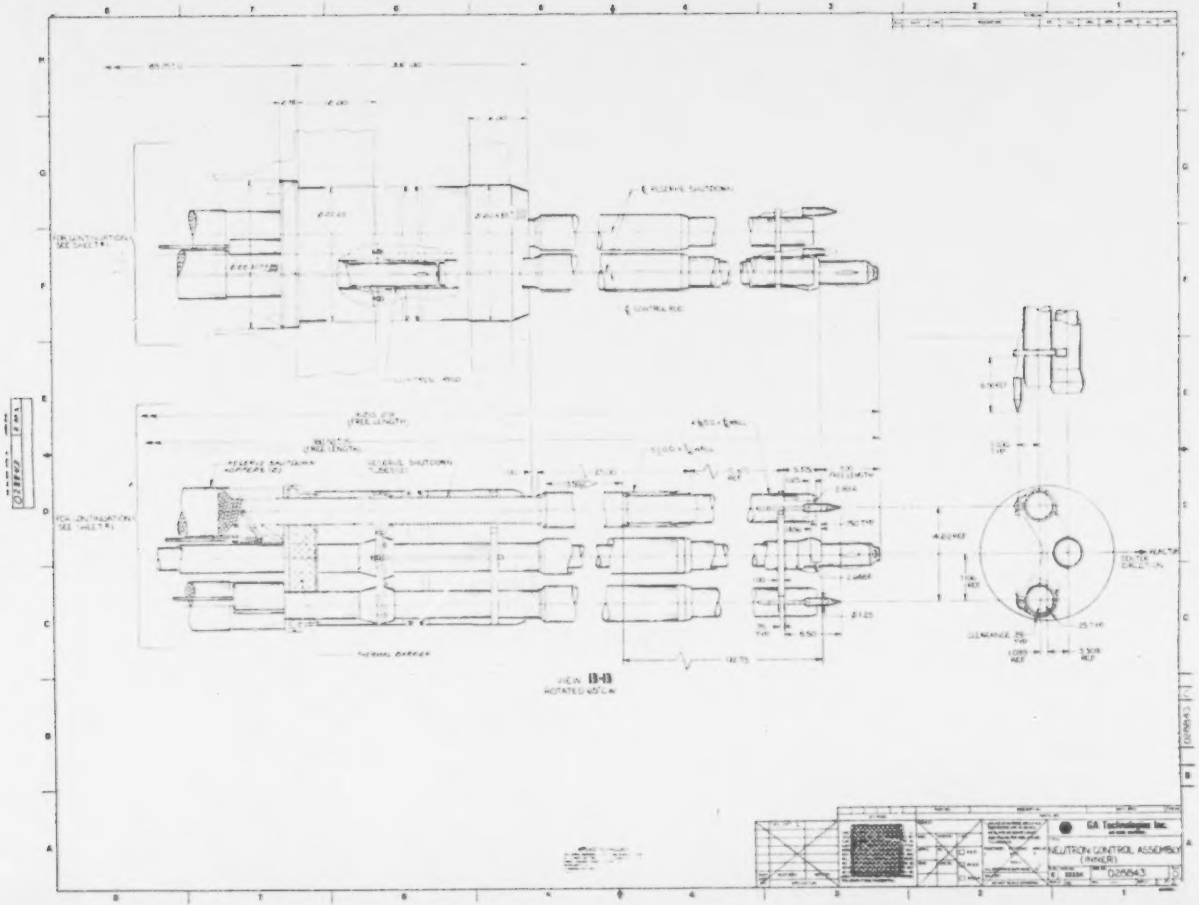
THIS TELESCOPING DIMENSION ALLOWS FOR
THERMAL BARRIER
NEUTRON SHIELD
GAMMA SHIELD
RESERVE SHUTDOWN RELEASE MECHANISM (2)
RESERVE SHUTDOWN FILLER PORTS (2)
VIEW B-B
ROTATED 45° CW
VIEW A-A
GA Technologies Inc.
NEUTRON CONTROL ASSEMBLY (INNER)
028843


GA Technologies Inc.
NEUTRON CONTROL ASSEMBLY
(INNER)
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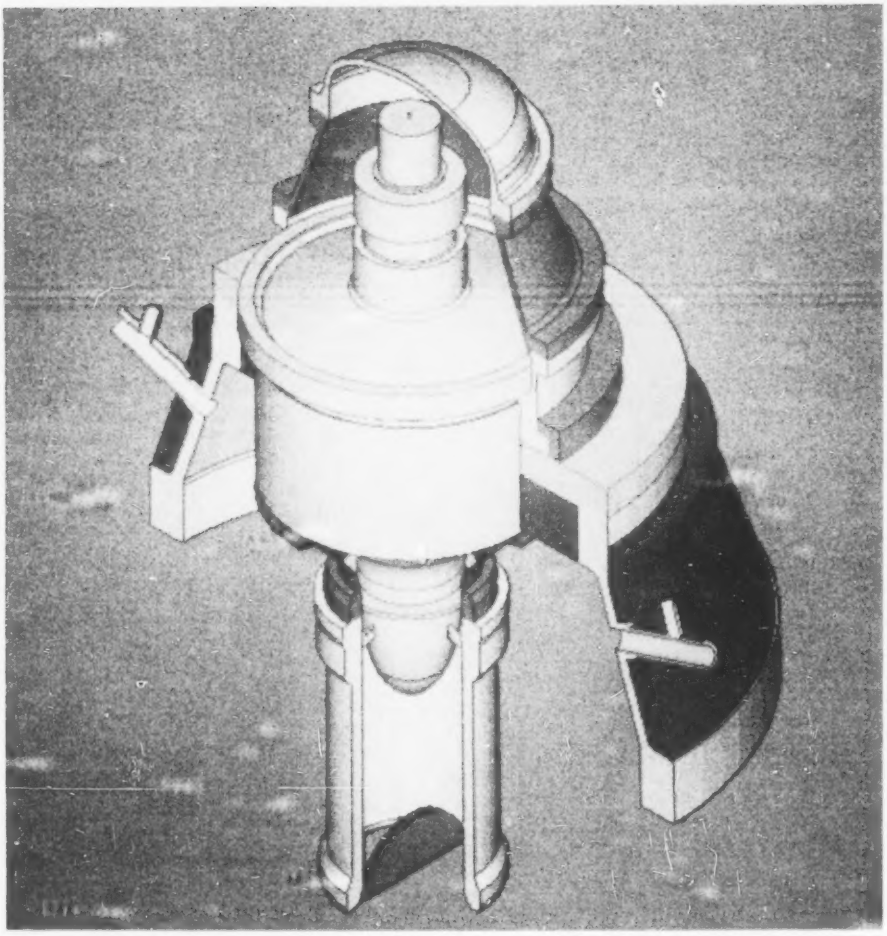
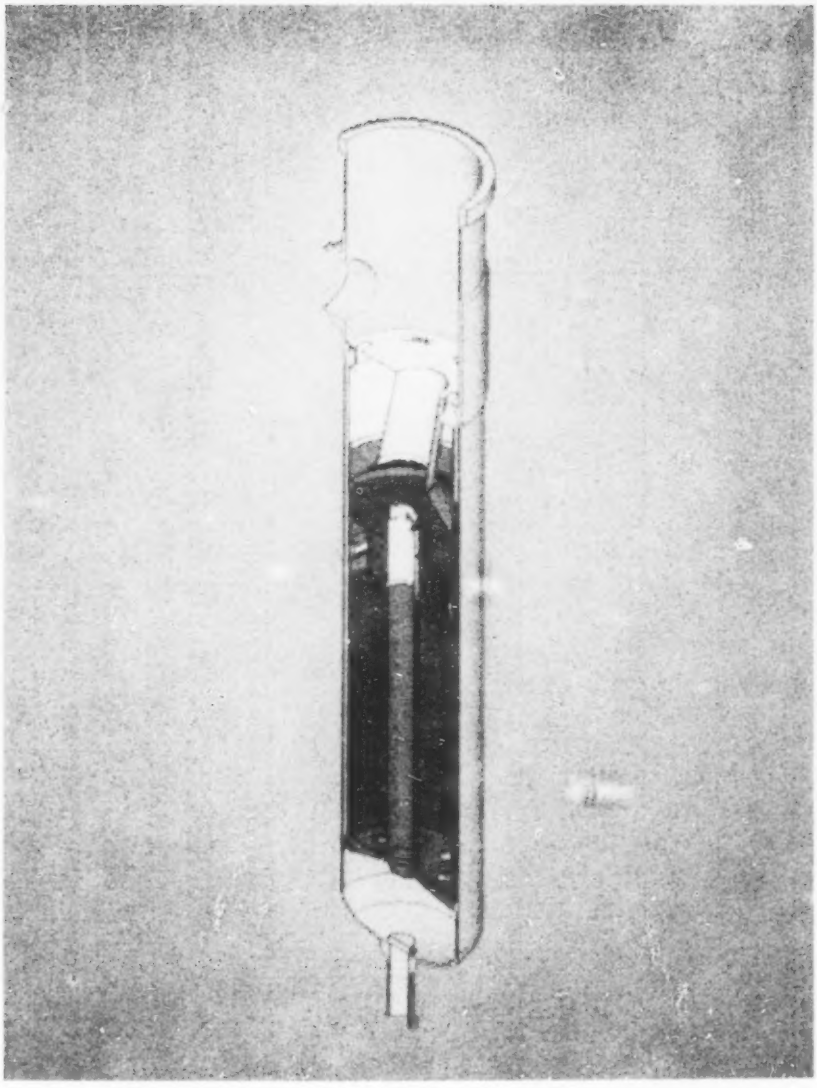
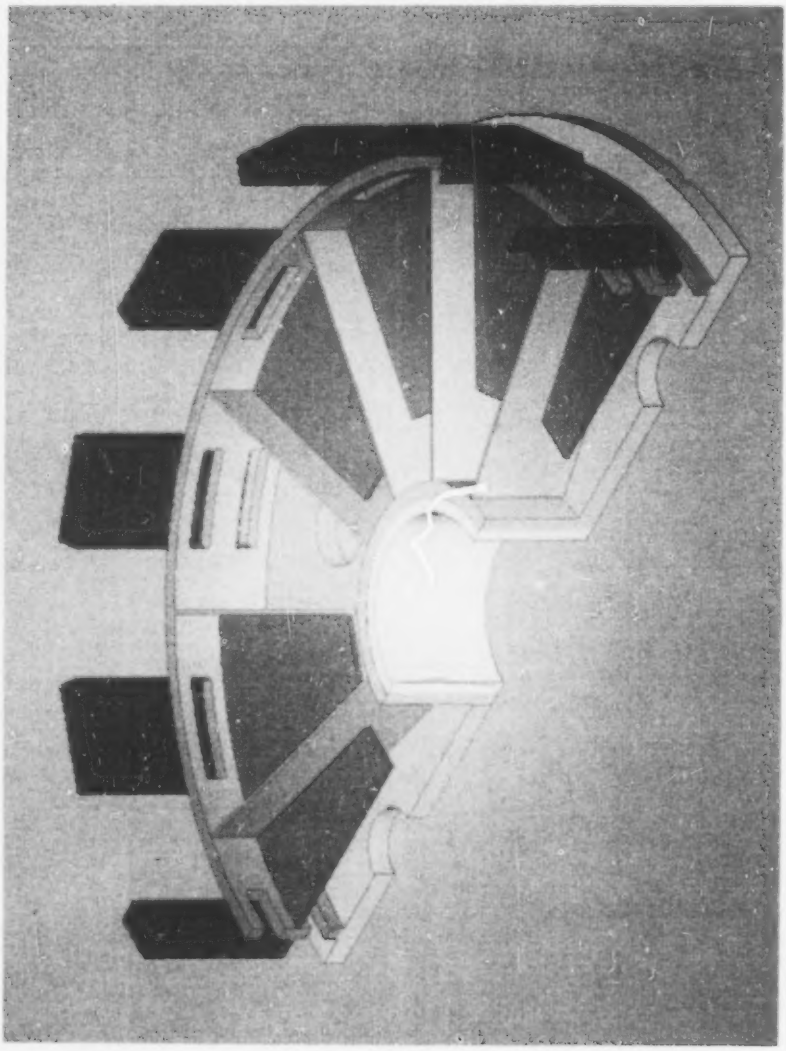
MAIN HELIUM CIRCULATOR

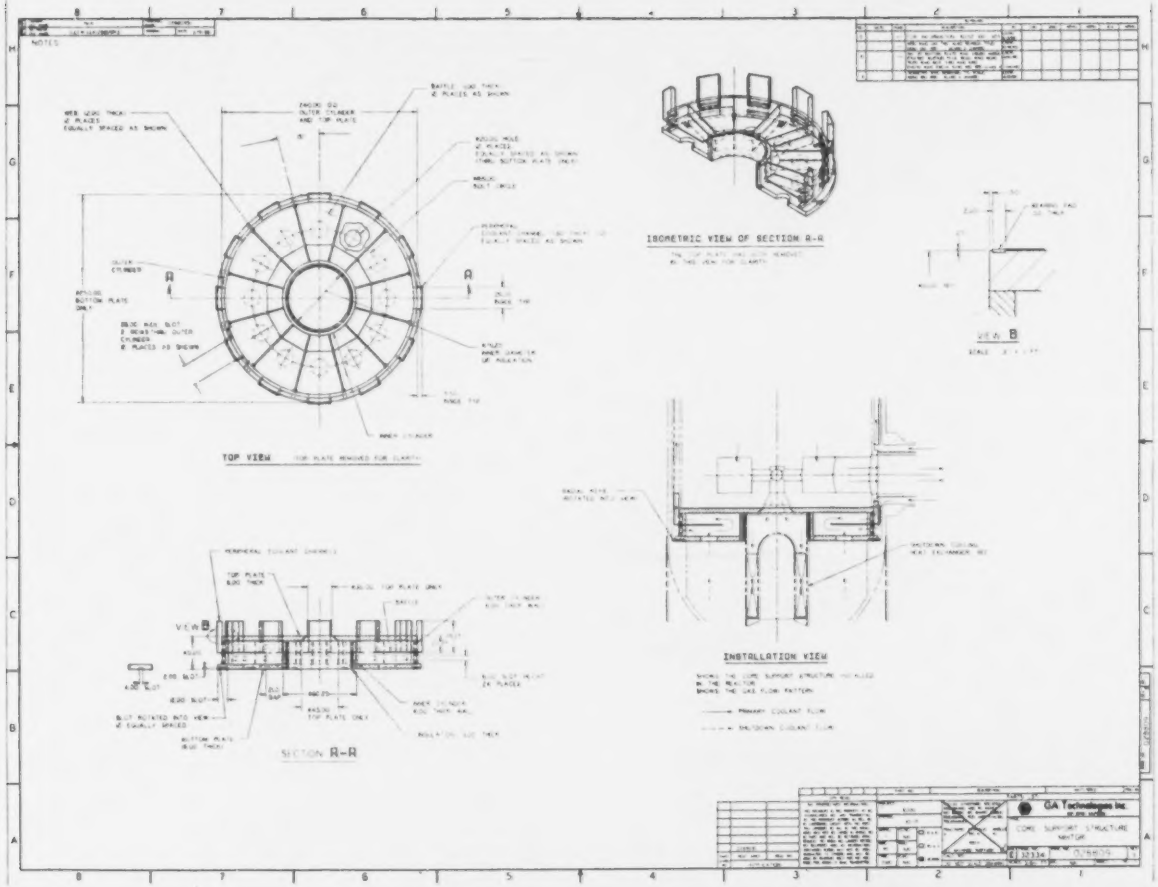


MAIN STEAM GENERATOR

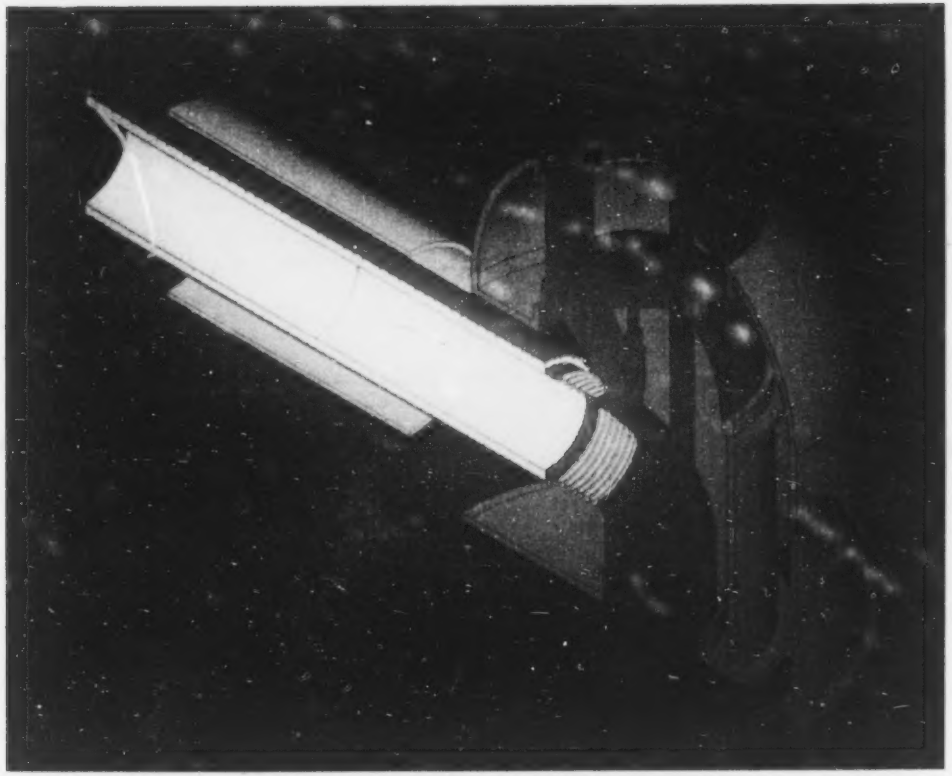


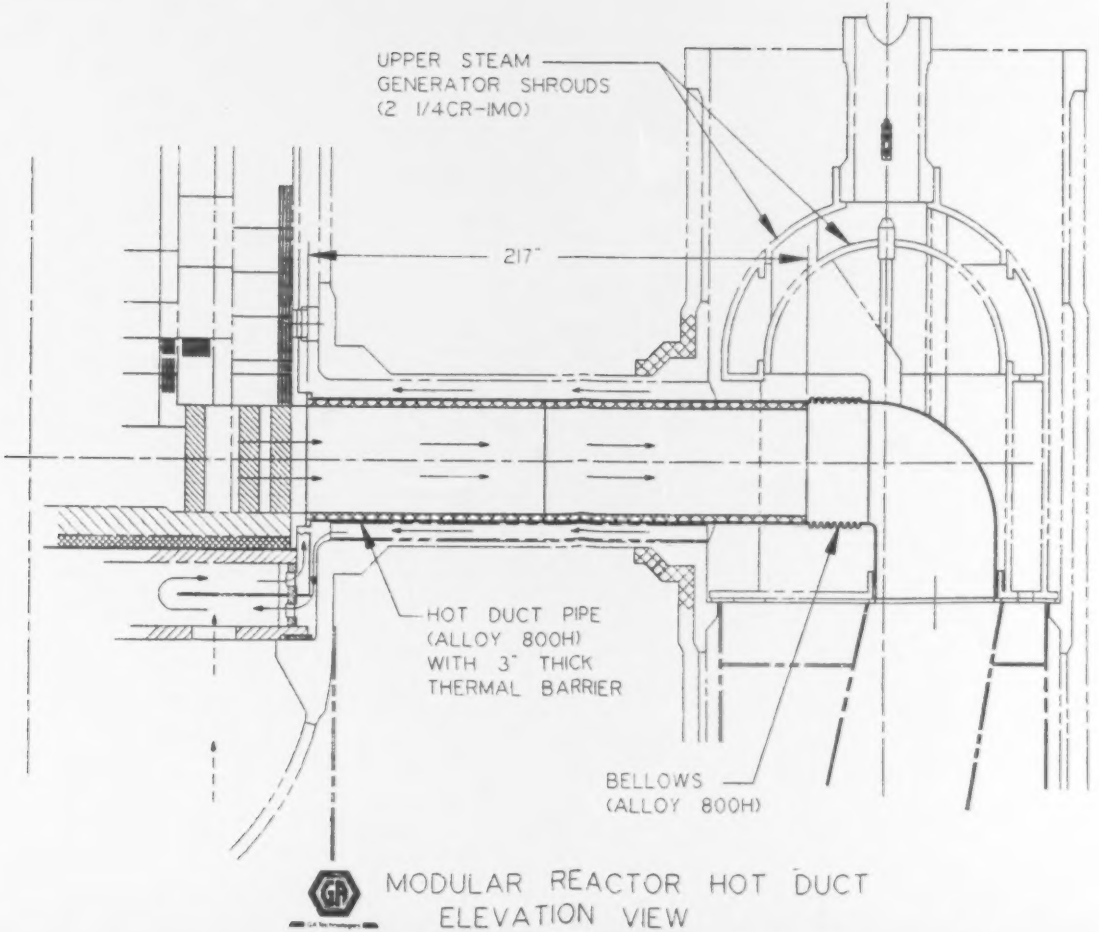
CORE SUPPORT FLOOR GRID



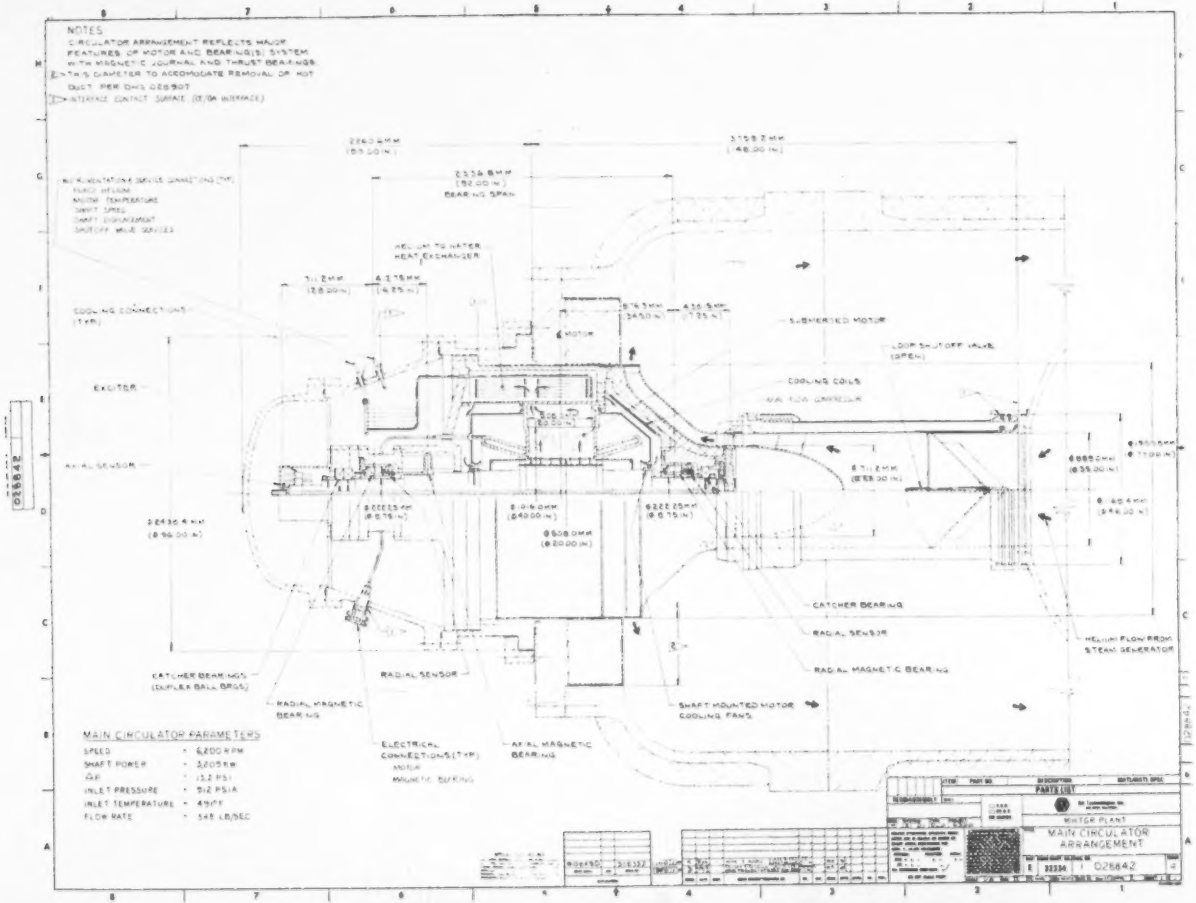

NOTES
TOP VIEW
ISOMETRIC VIEW OF SECTION R-R
VIEW B
SECTION R-R
INSTALLATION VIEW
GA Technologies Inc.
CORE SUPPORT STRUCTURE
MHTGR

CROSS DUCT TO STEAM GENERATOR

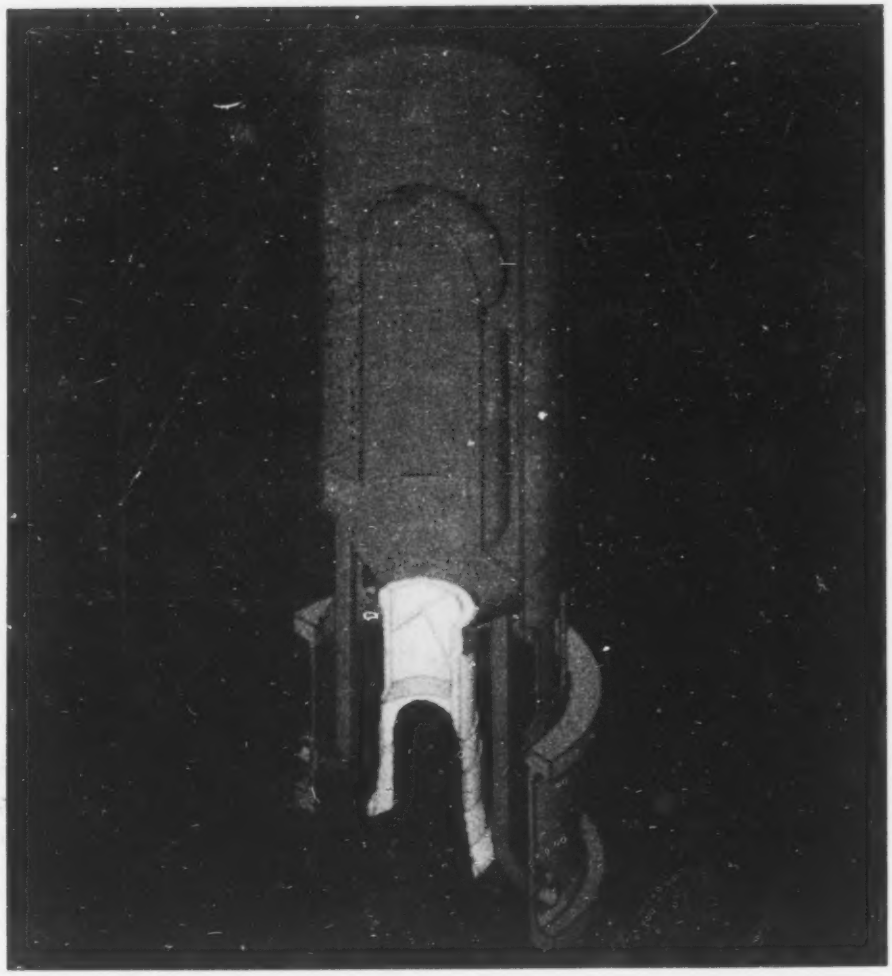


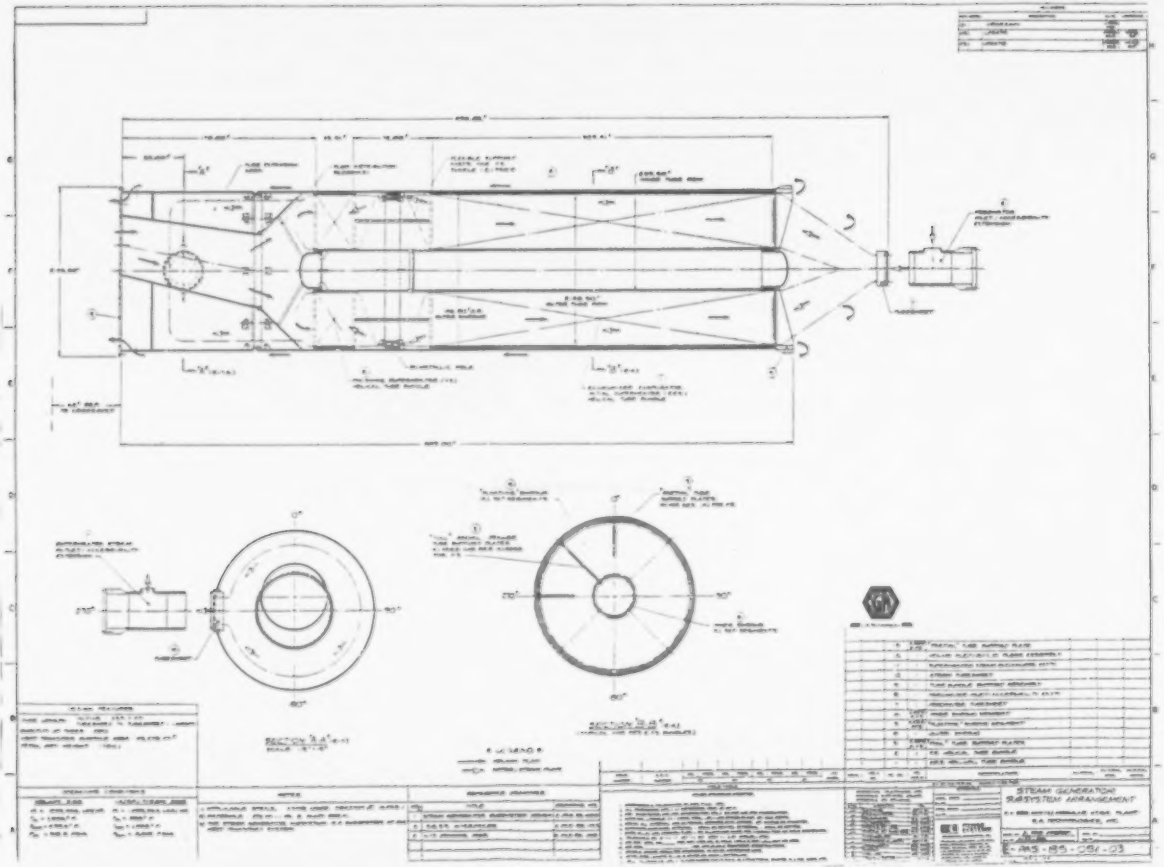


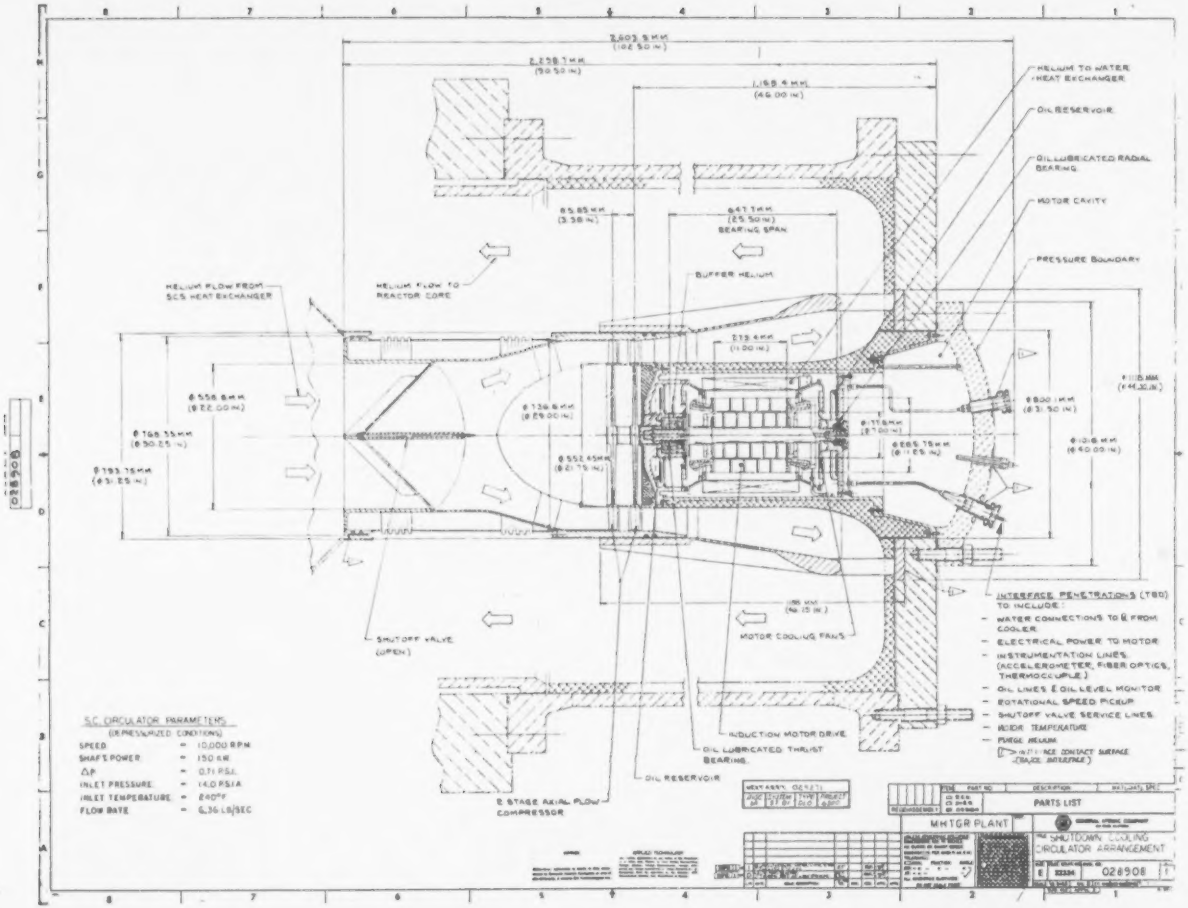
MODULAR REACTOR HOT DUCT
ELEVATION VIEW


NOTES
CIRCULATOR ARRANGEMENT REFLECTS MAJOR FEATURES OF MOTOR AND BEARING(S) SYSTEM WITH MAGNETIC JOURNAL AND THRUST BEARINGS.
2. 7 IN'S DIAMETER TO ACCOMODATE REMOVAL OF HOT DUCT PER DWG 026907
3. INTERFACE CONTACT SURFACE (0/OA INTERFACE)
INSTRUMENTATION DEVICE CONNECTIONS (TYP)
FLUID HELIUM
MOTOR TEMPERATURE
SHAFT SPEED
SHAFT DISPLACEMENT
SHUTOFF VALVE SERVICES
COOLING CONNECTIONS (TYP)
EXCITER
AXIAL SENSOR
HELIUM TO WATER HEAT EXCHANGER
SUBMERGED MOTOR
LOOP SHUTOFF VALVE (OPEN)
COOLING COILS
MAIN FLOW IMPELLER
CATCHER BEARING
RADIAL SENSOR
HELIUM FLOW FROM STEAM GENERATOR
RADIAL MAGNETIC BEARING
CATCHER BEARINGS (DUPLEX BALL BRGS)
RADIAL SENSOR
RADIAL MAGNETIC BEARING
SHAFT MOUNTED MOTOR COOLING FANS
ELECTRICAL CONNECTIONS (TYP) MOTOR MAGNETIC BUFFING
AXIAL MAGNETIC BEARING
MAIN CIRCULATOR PARAMETERS
SPEED - 6,200 RPM
SHAFT POWER - 3,205 KW
ΔP - 13.2 PSI
INLET PRESSURE - 512 PSIA
INLET TEMPERATURE - 491°F
FLOW RATE - 548 LB/SEC
PARTS LIST
WINTER PLANT
MAIN CIRCULATOR ARRANGEMENT
026842

SHUTDOWN CIRCULATOR AND HEAT EXCHANGER

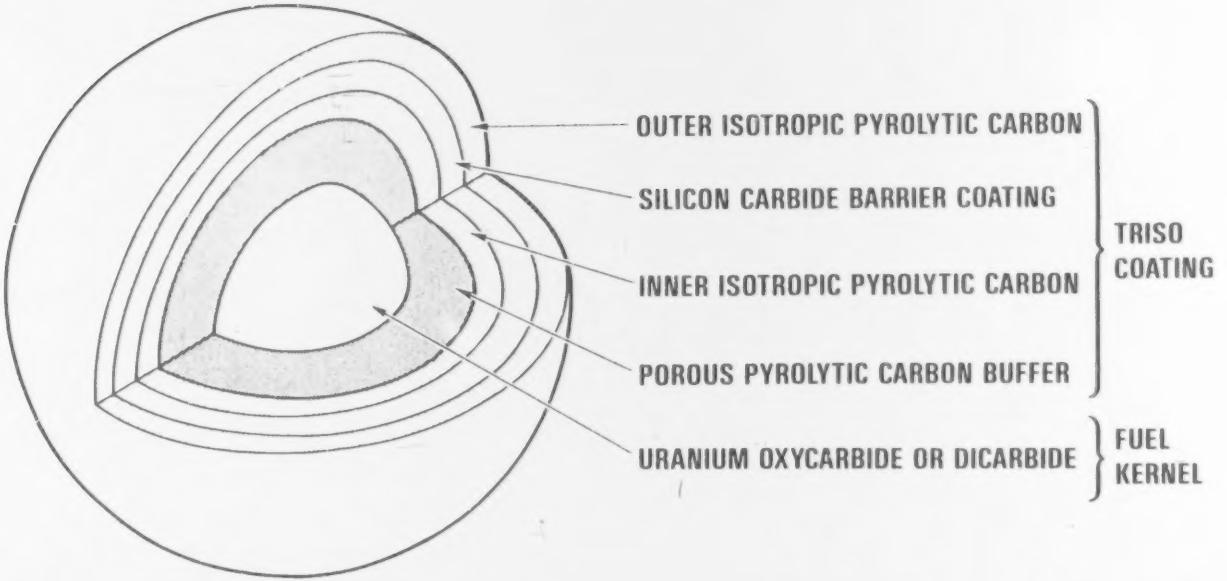




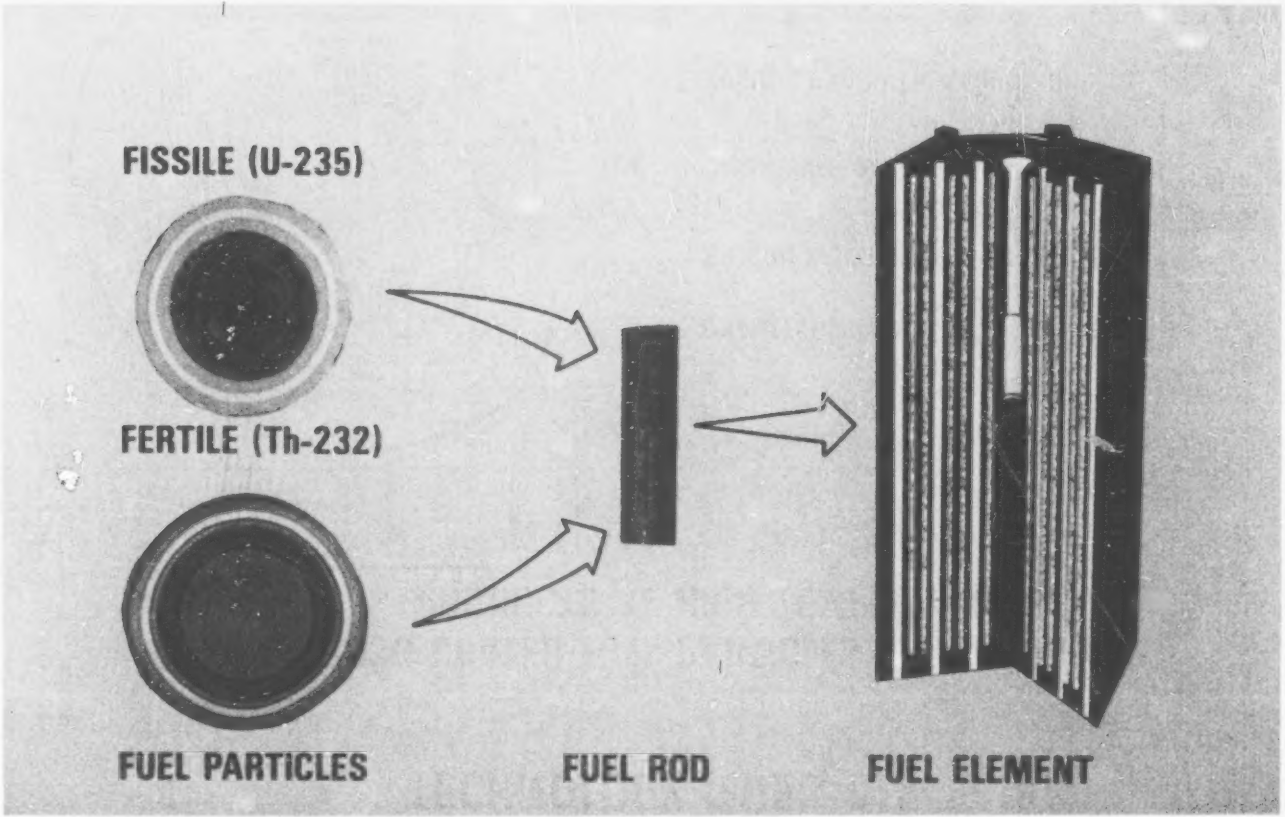

2603.5 MM (102.50 IN.)
2438.1 MM (96.00 IN.)
1168.4 MM (46.00 IN.)
HELIUM TO WATER HEAT EXCHANGER
OIL RESERVOIR
OIL LUBRICATED RADIAL BEARING
MOTOR CAVITY
85.85 MM (3.38 IN.)
647.7 MM (25.50 IN.) BEARING SPAN
PRESSURE BOUNDARY
HELIUM FLOW FROM SCS HEAT EXCHANGER
HELIUM FLOW TO REACTOR CORE
BUFFER HELIUM
279.4 MM (11.00 IN.)
Ø 558.8 MM (Ø 22.00 IN.)
Ø 736.6 MM (Ø 29.00 IN.)
Ø 800.1 MM (Ø 31.50 IN.)
Ø 1016 MM (Ø 40.00 IN.)
Ø 768.35 MM (Ø 30.25 IN.)
Ø 793.75 MM (Ø 31.25 IN.)
Ø 552.45 MM (Ø 21.75 IN.)
Ø 285.75 MM (Ø 11.25 IN.)
SHUTOFF VALVE (OPEN)
MOTOR COOLING FANS
INTERFACE PENETRATIONS (TBD) TO INCLUDE:
- WATER CONNECTIONS TO & FROM COOLER
- ELECTRICAL POWER TO MOTOR
- INSTRUMENTATION LINES (ACCELEROMETER, FIBER OPTICS, THERMOCOUPLE)
- OIL LINES & OIL LEVEL MONITOR
- ROTATIONAL SPEED PICKUP
- SHUTOFF VALVE SERVICE LINES
- MOTOR TEMPERATURE
- PURGE HELIUM
INTERFACE CONTACT SURFACE (BASIS: INTERFACE)
S.C. CIRCULATOR PARAMETERS (DEPRESSURIZED CONDITIONS)
SPEED = 10,000 RPM
SHAFT POWER = 150 kW
ΔP = 0.71 PSI.
INLET PRESSURE = 14.0 PSIA
INLET TEMPERATURE = 240°F
FLOW RATE = 6.36 LB/SEC
INDUCTION MOTOR DRIVE
OIL LUBRICATED THRUST BEARING
OIL RESERVOIR
2 STAGE AXIAL FLOW COMPRESSOR
PARTS LIST
MHTGR PLANT
SHUTDOWN COOLING CIRCULATOR ARRANGEMENT
026908

TECHNOLOGY (FUEL, MATERIAL)

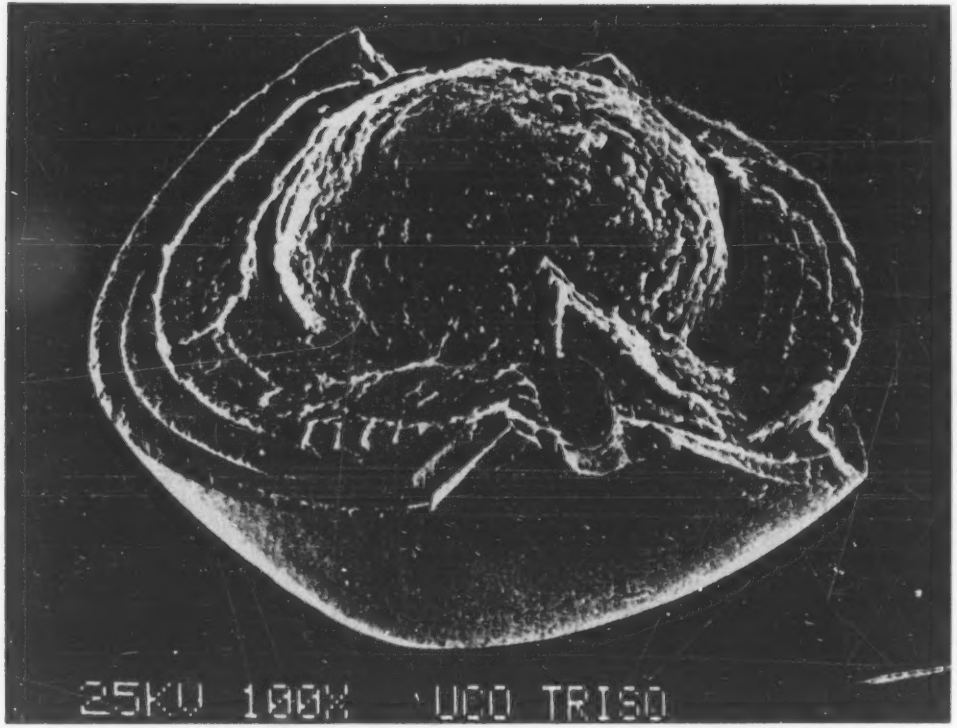
TRISO COATED FUEL PARTICLES RETAIN FISSION PRODUCTS AT HIGH TEMPERATURES



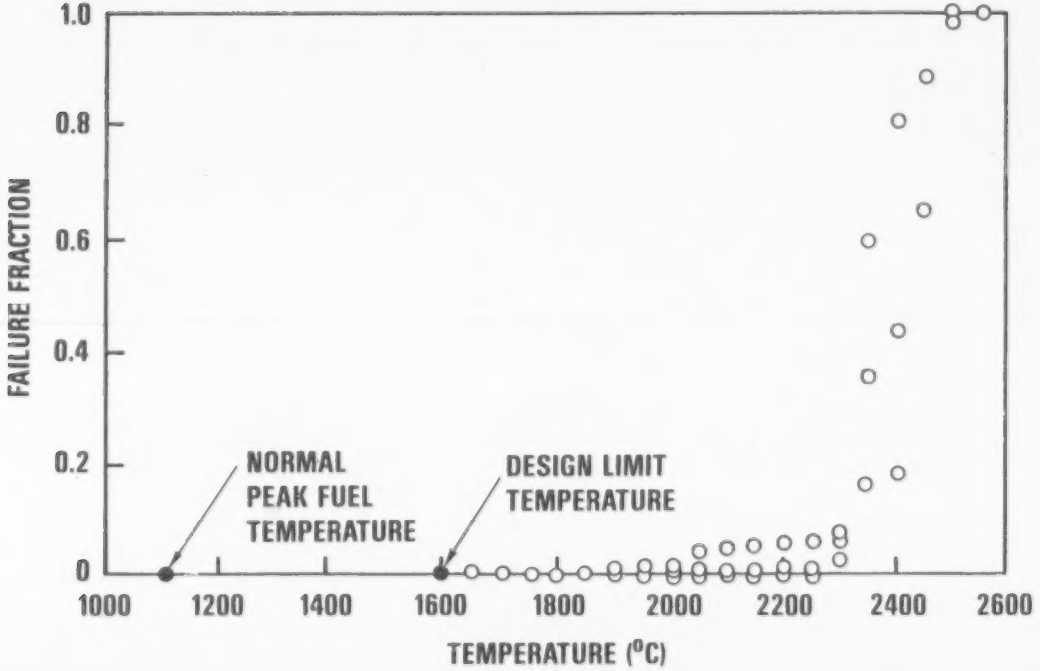
FUEL COMPONENTS



COATED PARTICLE W/DRILLED HOLE (BY LASER) THROUGH COATING

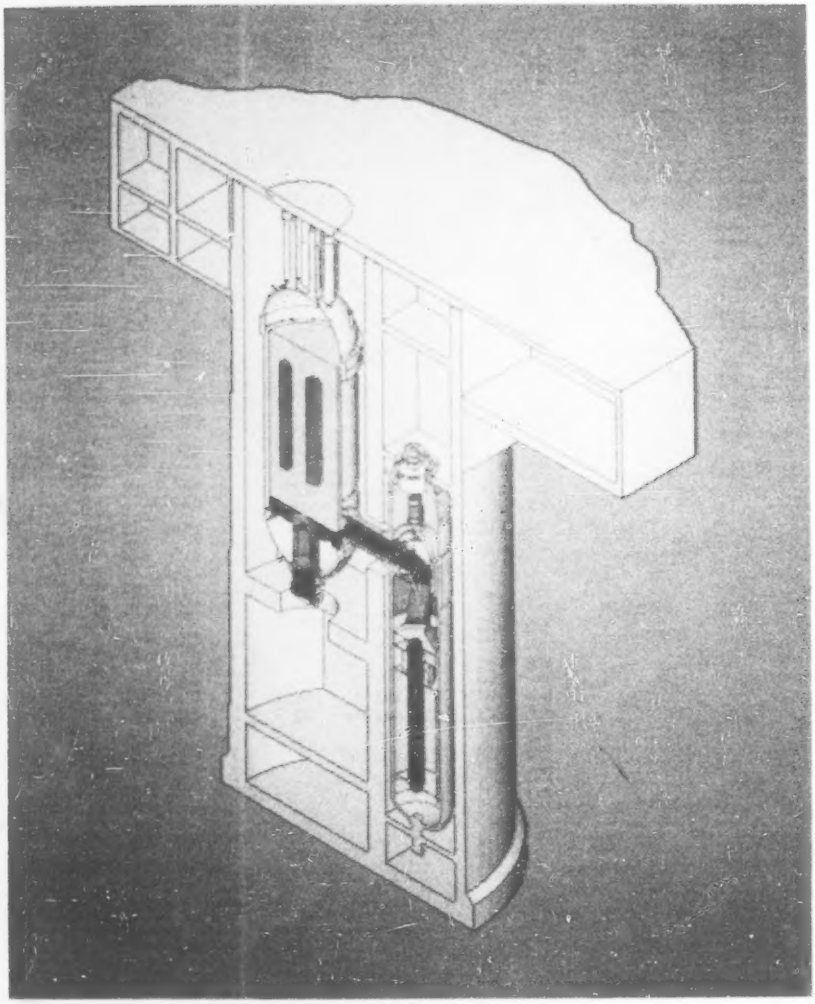


COATED FUEL PARTICLES MAINTAIN INTEGRITY AT HIGH TEMPERATURES



SAFETY ASSESSMENTS

SAFETY OF MODULAR HTGR WITH ANNULAR PRISMATIC CORE IN SILO



LICENSING INTERACTIONS

LICENSING INTERACTION SUMMARY ON MHTGR

MEETINGS WITH NRC 12

MEETINGS WITH ACRS 2

DOCUMENTS SUBMITTED
- LICENSING PLAN
- TOP LEVEL REGULATORY CRITERIA
- BRIDGING METHODS
- ACCIDENT SELECTION CRITERIA
- LICENSING BASIS EVENTS
- PRELIMINARY SAFETY INFORMATION DOCUMENT OUTLINE

DOCUMENTS BY END OF CY-86
- PSID
- PRA
- TECHNOLOGY PLAN

PROCEDURAL APPROACH INTERACTION

ADVANCED REACTOR POLICY	
– COMMISSION MEETING	11/83
– ISSUED FOR COMMENT	3/85
– INDUSTRY COMMENTS ON POLICY	5/85
– ADVANCED REACTOR POLICY ISSUED	7/86
MHTGR LICENSING PLAN	
– LICENSING PLAN DRAFT	10/84
– LICENSING PLAN FORMAL SUBMITTAL	4/85
– NRC ACCEPTANCE OF LICENSING PLAN	7/85
– SUSPENSION OF NRC INTERACTIONS	2/86
– RESUMPTION OF NRC INTERACTIONS	[9/86]

TECHNICAL APPROACH INTERACTION

TOP-LEVEL REGULATORY CRITERIA	
– BRIEFING	1/85
– SUBMITTAL	8/85
– NRC WRITTEN RESPONSE	7/86
– RESUBMITTAL FOR NRC APPROVAL	[10/86]
BRIDGING METHODS	
– BRIEFING ON BRIDGING METHODS	1/85
– BRIEFING ON ACCIDENT SELECTION CRITERIA	5/85
– BRIEFING ON FUNCTIONAL ANALYSIS	7/85
– BRIEFING ON SAFETY-RELATED SSC's	10/85
– BRIEFING ON PRINCIPAL DESIGN CRITERIA	10/85
– SUBMITTAL ON BRIDGING METHODS	7/86
– SUBMITTAL ON ACCIDENT SELECTION CRITERIA	7/86
– SUBMITTAL ON LICENSING BASIS EVENTS	7/86
– SUBMITTAL ON PRINCIPAL DESIGN CRITERIA	[10/86]

DESIGN/TECHNOLOGY FAMILIARIZATION

BRIEFINGS

– MODULAR HTGR	6/84
– HTGR TECHNOLOGY	8/84
– NO EVACUATION CRITERIA (NRC/FEMA)	11/84
– CONCEPT SELECTION	7/85
– REFERENCE DESIGN	12/85
– ACRS SUBCOMMITTEE	1/86
– FULL ACRS	3/86
– STANDARDIZATION	[9/86]
– ACCIDENTS/SOURCE TERMS	[10/86]
– CONTAINMENT SYSTEM	[11/86]

DESIGN & TECHNOLOGY REVIEW

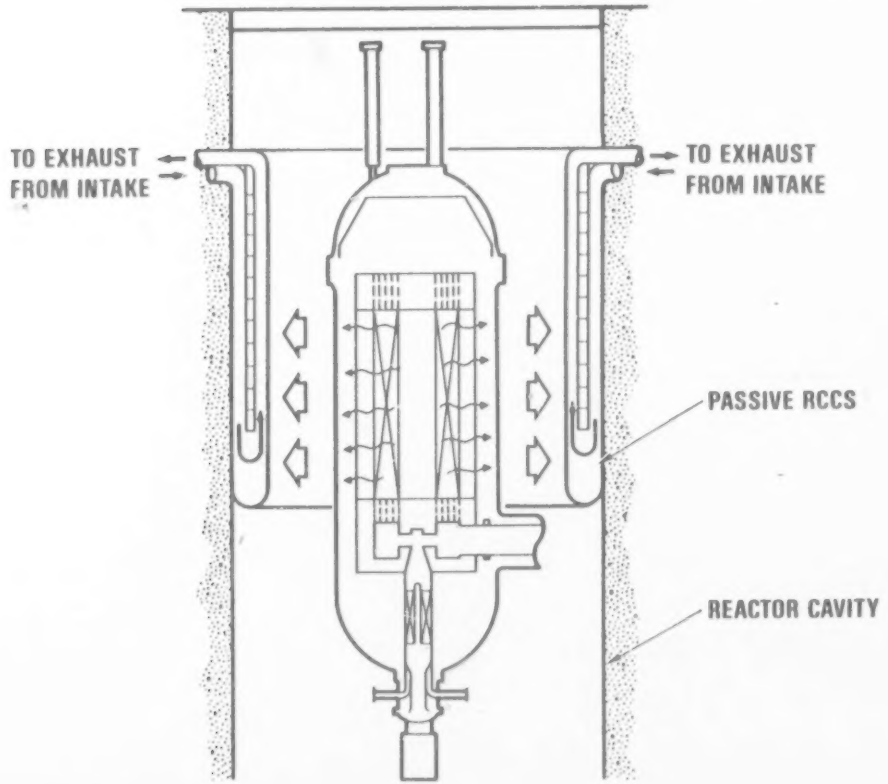
SUBMITTALS

– PSID OUTLINE	10/85
– PSID	[9/86]
– PRA	[12/86]
– TECHNOLOGY PLAN	[12/86]
– BASIS FOR EMERGENCY PLANNING	[2/87]

NRC REVIEW

– RESPONSE TO PSID OUTLINE	10/85
– DRAFT PRELIMINARY SAFETY EVALUATION REPORT	[10/87]
– FINAL PRELIMINARY SAFETY EVALUATION REPORT AND LICENSABILITY STATEMENT	[1/88]

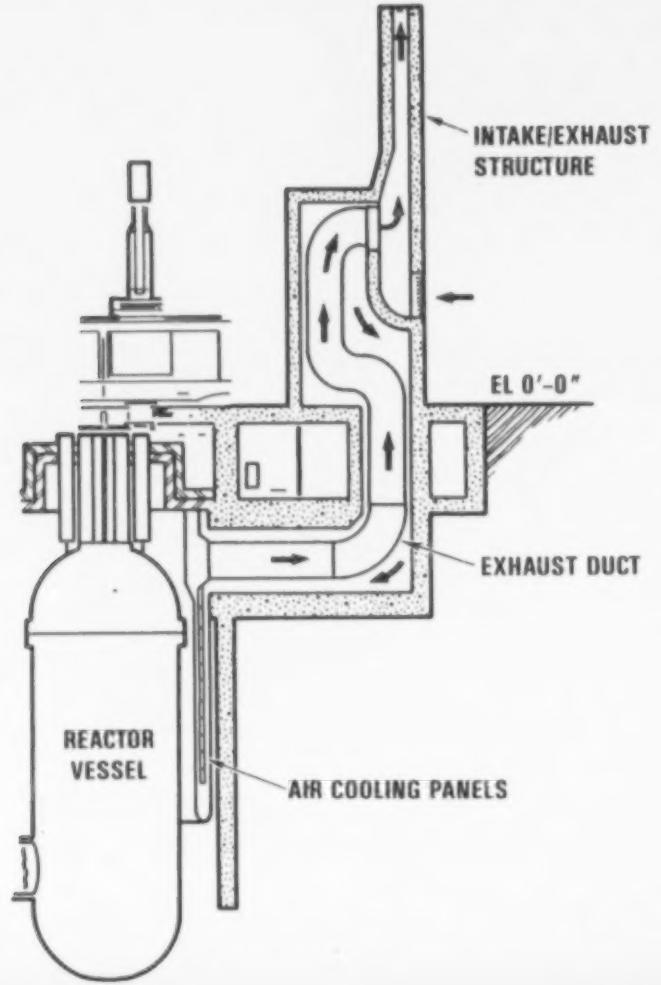
DEPRESSURIZED DECAY HEAT REMOVAL BY CONDUCTION AND RADIATION TO PASSIVE REACTOR CAVITY COOLING SYSTEM



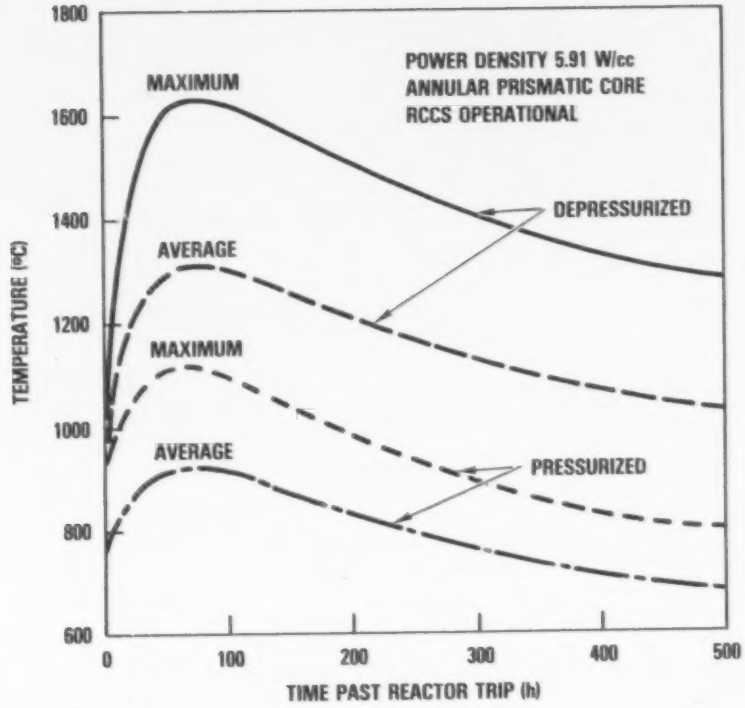
ESTIMATED PROJECT COST

	DOE Project Reference Plant + Demo Plant 1 x 350 MW(t)	Equilibrium Project 4 x 350 MW(t)
Cost Estimate (1/1986 $ Millions)		
Direct Capital Costs		
NSSS - Vessels	14	54
Fuel Handling	10	14
Steam Generator	6	23
Other	64	151
Nuclear Island	63	118
Energy Conversion Area		
Turbine Generator	17	64
Other	51	134
Total Direct Capital Costs	225	558
Indirect Capital Costs		
NSSS DES/SVS	204	
Nuclear Island DES/SVS	39	40
Energy Conv. Area DES/SVS	18	18
Construction Services	63	105
Total Indirect costs	324	163
Technology - NSSS	60	
Component Development	20	
Total Technology Development	80	
Owner's Cost	48	72
Contingency	70	132
Total Plant Capital Costs	463	925
Total Project Costs (Design + Capital	747	925
Cost/KWe (Dollars)	3307	1658

REACTOR CAVITY COOLING SYSTEM (ELEVATION)



350 MW(t) TRANSIENTS DURING DEPRESSURIZED AND PRESSURIZED CONDUCTION COOLDOWN



FUEL CYCLE REQUIREMENTS

FUEL CYCLE

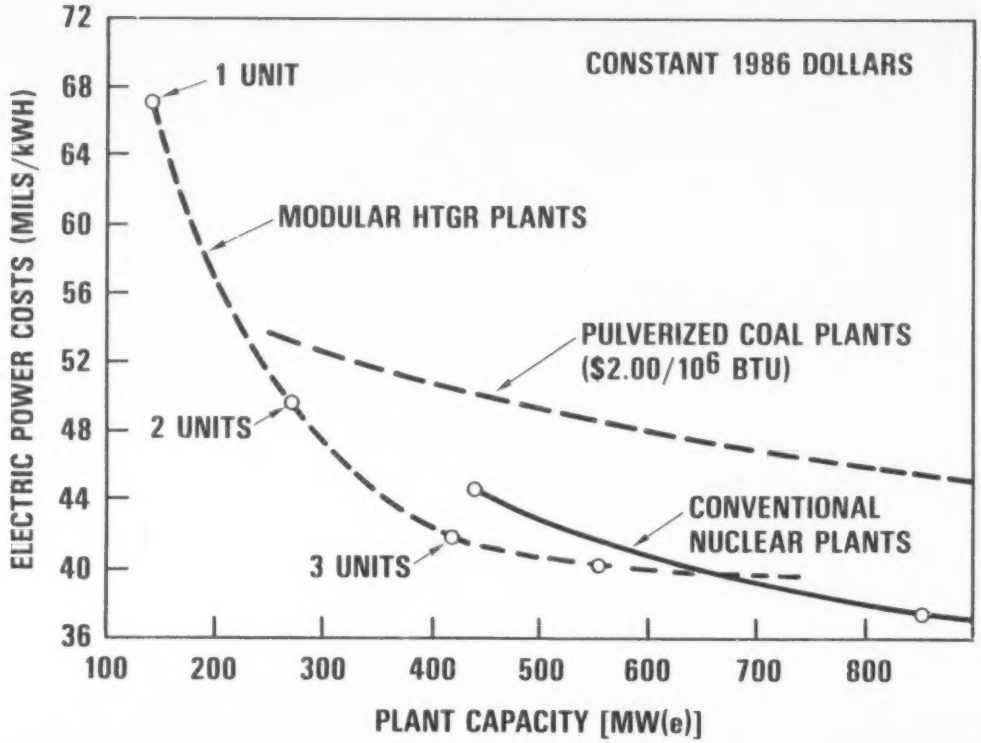
- LOW ENRICHED (< 20%) URANIUM-THORIUM ONCE-THROUGH FUEL CYCLE
- MEETS NATIONAL PROLIFERATION RESISTANCE POLICY
- NO REQUIREMENT TO CLOSE FUEL CYCLE
  - CERAMIC COATED FUEL PARTICLES EMBEDDED IN GRAPHITE AMENABLE TO LONG TERM STORAGE
  - OPTIONAL VOLUME REDUCTION TECHNIQUES AVAILABLE TO SEPARATE HIGH LEVEL FUEL RODS FROM LOW LEVEL GRAPHITE BLOCKS
- LOWER SPENT FUEL SOURCE LEVELS ALLOWS SHIPPING FROM SITES IN ONE VS. FIVE YEARS FOR LWR FUEL
- NO HIGH LEVEL LIQUID WASTES
- HEAT LOADS PER UNIT VOLUME OF STORAGE A FACTOR OF 2 TO 5 BELOW OTHER FUEL WASTE

ECONOMICS/COST ESTIMATE

PASSIVE SAFETY IS KEY TO MODULAR ECONOMICS

- TRADITIONAL SCALING LAWS FAVOR LARGER PLANT
- MODULAR HTGR ECONOMICS BENEFIT FROM
  - ELIMINATION OF EXCESS ENGINEERED SAFETY SYSTEMS
  - NONSAFETY CLASSIFICATION OF BALANCE OF PLANT
  - PRE-LICENSED, STANDARDIZED DESIGN
  - SHORTENED SCHEDULES
  - SERIALIZED FACTORY FABRICATION
  - MULTIPLE UNITS ON A SITE
- EARLY ECONOMIC RESULTS POSITIVE

MODULAR HTGR POWER COSTS MEET THE ECONOMIC GOALS



MARKET INTEREST/PRIVATE SECTOR SUPPORT

GAS–COOLED REACTOR ASSOCIATION
UTILITY GROUP SUPPORTING MHTGR

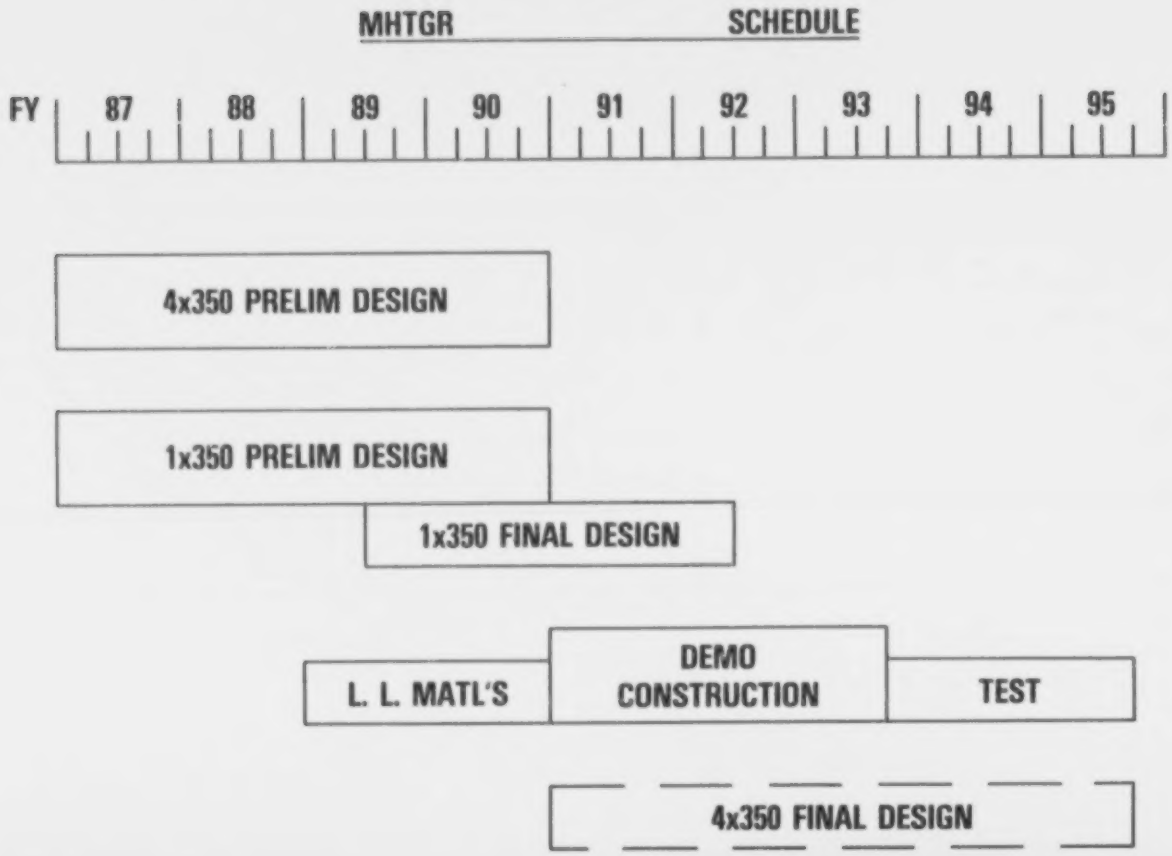
Arizona Public Service Company
Baltimore Gas & Electric Company
Colorado–Ute Electric Association
Delmarva Power & Light Company
Empire State Electric Energy Resource Corp.
- Central Hudson Gas & Electric
- Consolidated Edison Company
- New York Power Authority
- New York State Electric & Gas
- Niagara Mohawk Power
- Orange & Rockland Utilities
- Rochester Gas & Electric Corp.

Florida Power & Light Company
Gulf States Utilities
Idaho Power Company
Northeast Utilities
- Connecticut Light & Power
- Western Massachusetts Electric
- Holyoke Water Power Company

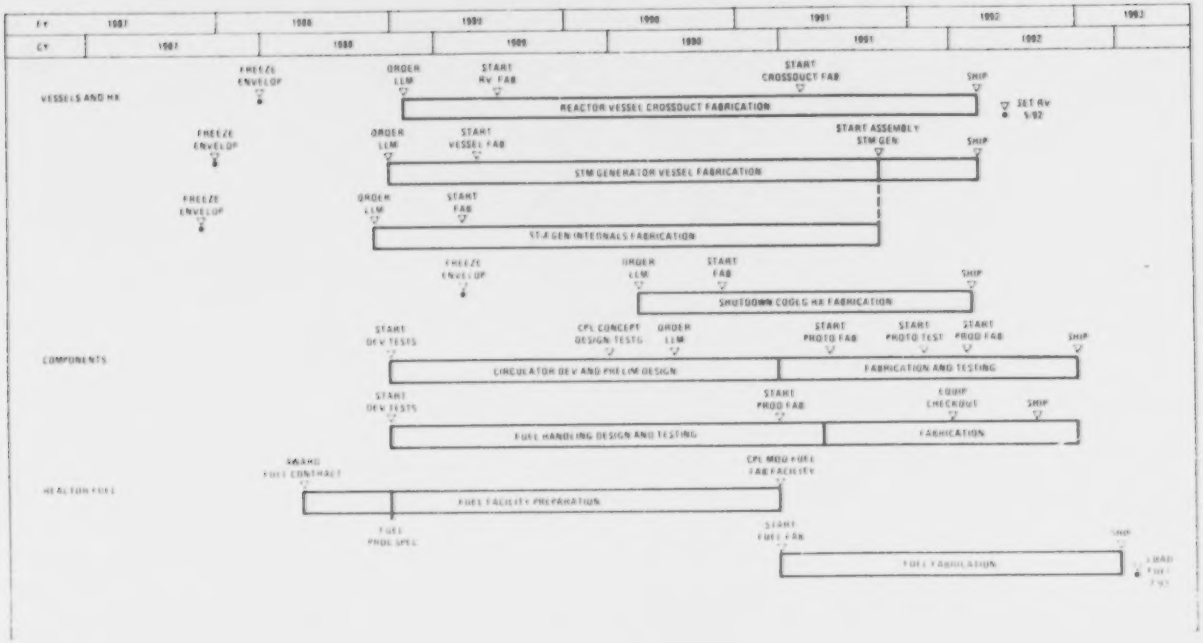
Pacific Gas & Electric Company
Pacific Power & LIght Company
Philadelphia Electric Company
Public Service Company of Colorado
Public Service Company of New Mexico
Public Service Electric & Gas Company
Salt River Project
San Diego Gas & Electric
Tennessee Valley Authority
Yankee Atomic Electric Company

In addition, GCRA's constituency includes potential process heat users who are interested in applying the HTGR in their energy–intensive industries in the future.

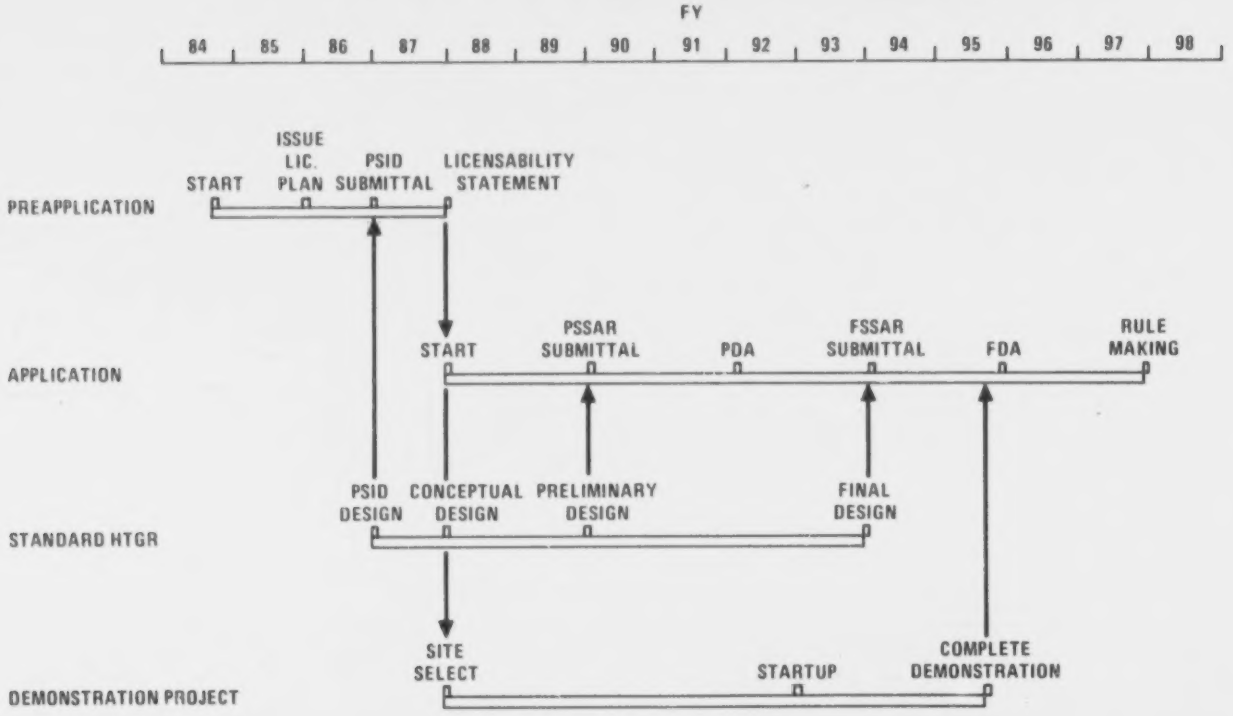
SCHEDULE


MHTGR SCHEDULE
FY
87
88
89
90
91
92
93
94
95
4x350 PRELIM DESIGN
1x350 PRELIM DESIGN
1x350 FINAL DESIGN
L. L. MATL'S
DEMO CONSTRUCTION
TEST
4x350 FINAL DESIGN

LONG LEAD ORDERS FOR PROJECT COMPONENTS



SCHEDULE FOR HTGR LICENSING ACTIVITIES



INTERNATIONAL COOPERATION

USA–FRG

- GOVERNMENT LEVEL:
  FUEL, FISSION PRODUCTS, GRAPHITE, METALS AND SAFETY RESEARCH
- INDUSTRY TO INDUSTRY:
  REACTOR DESIGN & FUEL MANUFACTURE
- UTILITY TO UTILITY:
  USER REQUIREMENT & PROJECT DEVELOPMENT PLANNING

USA–JAPAN

- GOVERNMENT LEVEL:
  FUEL, GRAPHITE, METALS, REACTOR COMPONENT TESTS
- INDUSTRY TO INDUSTRY:
  INDEPENDENT DESIGN REVIEWS

USA–UK

- UTILITY & INDUSTRY:
  REACTOR OPERATING EXPERIENCE

USA–FRANCE

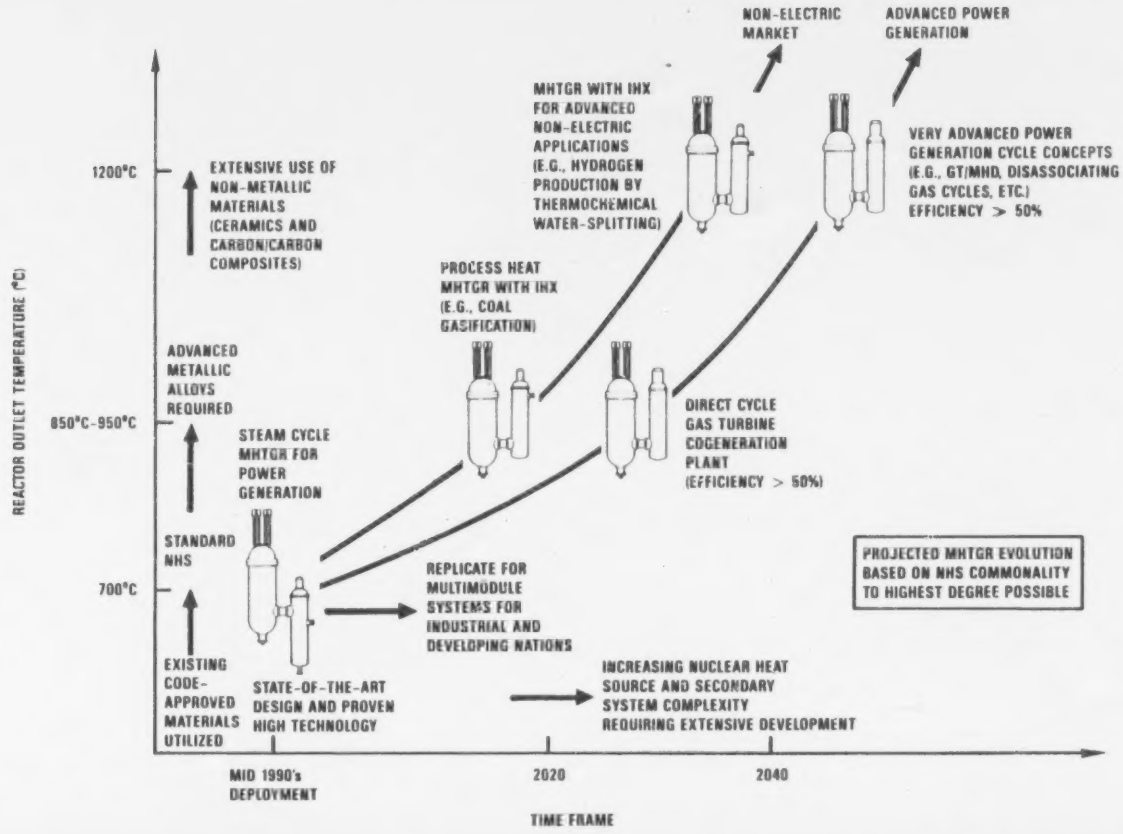
- PRIOR INDUSTRY EXCHANGE COMPLETED ON FUEL AND MAJOR COMPONENT TESTING
- GOVERNMENT LEVEL:
  FISSION PRODUCT RESEARCH BEING PLANNED

ADVANCED APPLICATIONS OF MHTGR

HTGR TECHNOLOGY PROVIDES OTHER POWER SYSTEM OPTIONS

- MODULAR HTGR–GT
  - CLOSED CYCLE GAS TURBINE
  - EFFICIENCIES OVER 40%
- MODULAR HTGR–PH
  - VERY HIGH TEMPERATURES FOR PROCESS HEAT
  - HYDROGEN PRODUCTION
- INTEGRATED HTGR
  - LARGE [440 MW(e) TO 1300 MW(e)] UNIT SIZES
  - NUCLEAR SYSTEMS INTEGRATED IN PRESTRESSED CONCRETE REACTOR VESSEL

MODULAR HTGR DEPLOYMENT STRATEGY


NON-ELECTRIC MARKET
ADVANCED POWER GENERATION
MHTGR WITH IHX FOR ADVANCED NON-ELECTRIC APPLICATIONS (E.G., HYDROGEN PRODUCTION BY THERMOCHEMICAL WATER-SPLITTING)
VERY ADVANCED POWER GENERATION CYCLE CONCEPTS (E.G., GT/MHD, DISASSOCIATING GAS CYCLES, ETC.) EFFICIENCY > 50%
1200°C
EXTENSIVE USE OF NON-METALLIC MATERIALS (CERAMICS AND CARBON/CARBON COMPOSITES)
PROCESS HEAT MHTGR WITH IHX (E.G., COAL GASIFICATION)
ADVANCED METALLIC ALLOYS REQUIRED
850°C-950°C
STEAM CYCLE MHTGR FOR POWER GENERATION
DIRECT CYCLE GAS TURBINE COGENERATION PLANT (EFFICIENCY > 50%)
STANDARD NHS
700°C
REPLICATE FOR MULTIMODULE SYSTEMS FOR INDUSTRIAL AND DEVELOPING NATIONS
PROJECTED MHTGR EVOLUTION BASED ON NHS COMMONALITY TO HIGHEST DEGREE POSSIBLE
EXISTING CODE-APPROVED MATERIALS UTILIZED
STATE-OF-THE-ART DESIGN AND PROVEN HIGH TECHNOLOGY
INCREASING NUCLEAR HEAT SOURCE AND SECONDARY SYSTEM COMPLEXITY REQUIRING EXTENSIVE DEVELOPMENT
REACTOR OUTLET TEMPERATURE (°C)
MID 1990's DEPLOYMENT
2020
2040
TIME FRAME

APPENDIX A

MHTGR - CONCEPTUAL DESIGN DATA

1. INTRODUCTION

This appendix provides the basic design parameters for the Modular HTGR concept. These design parameters are consistent with those presented in the Overall Plant Design Specification (OPDS) and the System Design Description (SDD) documents.

2. GENERAL PLANT PARAMETERS

a.	Plant power rating, MW(t)	1400
b.	No. modules	4
c.	Plant power operating range	25-100%
d.	Type of pressure vessel	Steel
e.	Type of fuel	Prismatic elements
f.	Core configuration	Annular
g.	Mode of emergency cooling	Passive
h.	Reactor enclosure	Reactor building

3. SITE CONDITIONS

a.	EAB radius	425 m
b.	LPZ radius	1600 m
c.	OBE/SSE	0.15/0.30 g
d.	Soil	Shear wave Vel = 1000-1800 fps, 10KSF bearing
e.	Ambient (dry bulb) temperatures (max, min)	110°F/-45°F
f.	Ambient (web bulb) temperatures (max)	82°F

3.1 NHS PARAMETERS

3.1.1 Reactor Vessel

a.	Material of construction	Mn-Mo;SA-533, Grade B, Class 1
b.	Operating temperature, °C/°F	205/400
c.	Operating pressure, MPa/psia	6.93/925
d.	Height, w/CRDs and shutdown circulator, m/ft	29.42/28.93/94.9
e.	Vessel, (nominal) o.d., m/ft	6.81/22.35
f.	Vessel, (nominal) thickness, cm/in.	13.34/5.25
g.	Support location	Bottom of vessel
h.	Vessel, weight, ton	750
i.	Number and location of refueling penetrations	6, Vessel top head

APPENDIX A (cont)

MHTGR - CONCEPTUAL DESIGN DATA

3.1.2 Steam Generator Vessel/Circulator Closure

a.	Material of construction	Mn-Mo; SA-533, Grade B
b.	Operating temperature, °C/°F	255/490
c.	Operating pressure, MPa/psia	6.93/925
d.	Height, m/ft (from steam generator closure flange to FW nozzle)	26.45/86.8
e.	Vessel, o.d., m/ft	4.18/13.71
f.	Vessel, thickness, cm/in.	8.89/3.50
g.	Support location	At cross duct
h.	Vessel weight, ton	308.1
i.	Tube bundle weight, ton	208.1
j.	Height of circulator closure, m/ft	2.50/8.19
k.	Number of pressure relief penetrations(1)	2

3.1.3 Reactor Vessel-Steam Generator/Cross Duct (Insulated)

a.	Material of construction - outer duct	Mn-Mo; SA-533, Grade B
b.	Operating temperature, °C/°F - outer duct	260/500
c.	Operating pressure, MPa/psia - outer duct	6.93/925
d.	Material of construction - inner duct	ASME SB409
e.	Operating temperature, °C/°F - inner duct	688/1270
f.	Operating pressure, MPa/psia - inner duct	6.33/918
g.	Length (vessel o.d. to vessel o.d.), m/ft	4.26/13.97
h.	Outer wall o.d., m/ft	1.93/6.33
i.	Outer wall thickness, cm/in.	6.35/2.5
j.	Inner wall o.d., m/ft	1.37/4.50
k.	Inner wall thickness, cm/in.	1.27/0.50
l.	Thermal barrier thickness(2), cm/in.	7.62/3.0
m.	Size of pressure relief penetrations, cm/in.	5.08/2
n.	Location of pressure relief penetrations	Top of steam generator vessel (180° apart)

3.1.4 Core Design

a.	Inner diameter - active core, m/ft	1.65/5.41
b.	Outer diameter - active core, m/ft	3.50/11.48
c.	Height - active core, m/ft	7.93/26.02
d.	Outer radial reflector thickness, m/ft	1.0/3.28
e.	Axial reflector height - top, m/ft	1.59/5.22
f.	Axial reflector height - bottom, m/ft	1.79/5.87
g.	Power density, w/cc	5.91
h.	Control rod location	Top mounted

(1) Pressure Relief protects both steam generator and reactor vessels.

(2) Includes coverplate = 1.27 cm/0.5 in.

APPENDIX A (cont)

MHTGR - CONCEPTUAL DESIGN DATA

i.	Number of columns in active core	66
j.	Number of fuel elements in active core	660
k.	Number of control rods	24 in outer reflector 6 in inner reflector
l.	Reserve shutdown system	12 in inner core row
m.	Refueling method(3)	2-segment, graded cycle(4)
n.	He temp @ core outlet, rated power, °C/°F	687/1268
o.	He temp @ core inlet, rated power, °C/°F	258/497
p.	Core inlet pressure, MPa/psia	6.37/924
q.	Core outlet pressure, MPz/psia	6.34/919
r.	Pressure drop across active core, kPa/psid	34.5/5.0
s.	Core circulation direction	Down Flow
t.	Coolant volume fraction	0.19
u.	Shielding	Boronated graphite
v.	Core support	Bottom
w.	Fuel loading,(5) initial core: kg U/kg Th	1394/2354
x.	Fuel loading,(5) equilibrium reloads: kg U/kg Th	965/881
y.	Lateral restraint configuration	Core barrel with seismic keys

3.1.5 Main Circulator

a.	Number of circulators	1
b.	Circulator drive type	Submerged electric motor, variable speed
c.	Location	Top of steam generator vessel
d.	Circulator drive power, kW	3,600
e.	He press. @ circulator discharge, MPa/psia	6.38/925
f.	Circulator flow rate, kg/hr/lb/hr	569,000/1,254,000
g.	Circulator pressure rise, kPa/psid	91.0/13.2
h.	Circulator type	Single stage axial
i.	Circulator bearings	Magnetic
j.	Helium shutoff valve	At entrance to circulator

(3) Shutdown for ~TBD days at end of cycle; all cycles: 1-1/2 years.

(4) Half of core replaced after each reload interval.

(5) LEU/Th; uranium (19.9% enriched in U-235) plus thorium.

APPENDIX A (cont)

MHTGR - CONCEPTUAL DESIGN DATA

3.1.6 Steam Generator

a.	Location	Bottom centerline of SG/Circ vessel
b.	Boiling direction	Uphill
c.	Number of SGs, thermal power each	1 - 352.15 MW(t)
d.	Steam temp at SG outlet, °C/°F	541/1005
e.	Steam press at SG outlet, MPa/psia	17.34/2515
f.	Feedwater temperature, °C/°F	193/380
g.	Feedwater pressure, MPa/psia	20.68/3000
h.	Pressure drop (He side), kPa/psid	25.5/3.7
i.	Type	Once through, cross counterflow helical coil
j.	FW penetration location	Bottom of vessel
k.	Steam penetration location	Near mid-plane of vessel
l.	Tube material	Alloy 800H (SH) & 2-1/4 Cr-1 Mo (EES)
m.	Outer shroud, o.d., m/ft	3.93/12.9
n.	Tube bundle, height, m/ft	9.72/31.9
o.	Type of S/G blowdown sink	Blowdown tanks

3.1.7 Shutdown Cooling Circulator

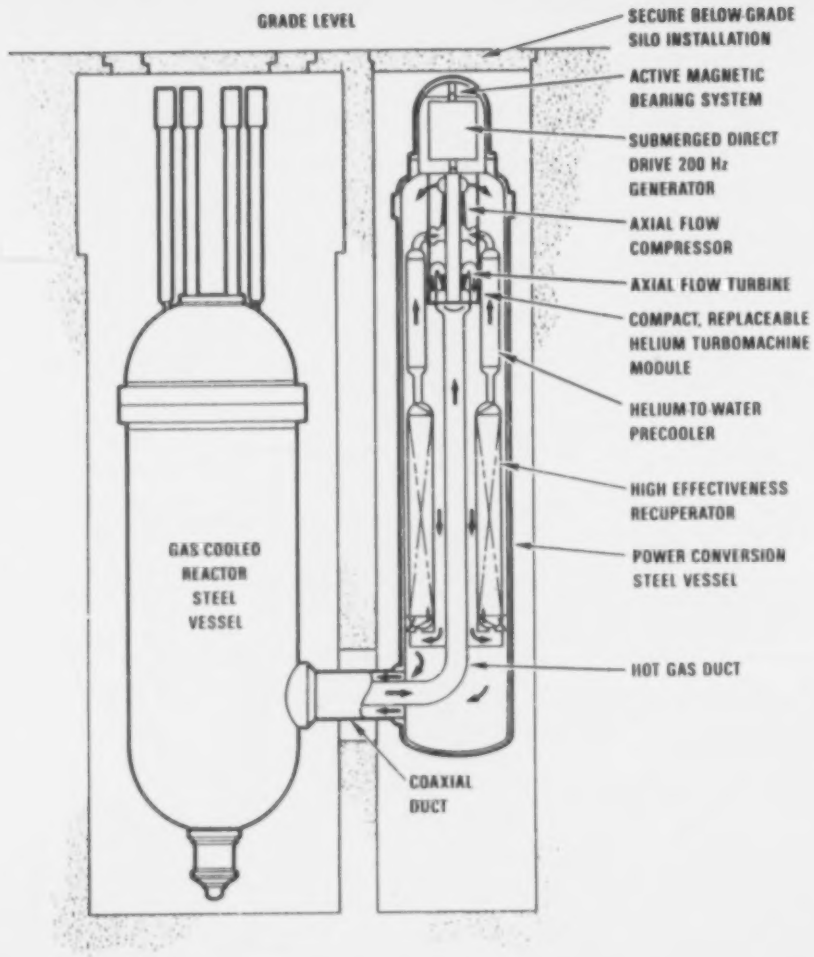
a.	Number of shutdown circulators	1
b.	Circulator drive	Electric motor, variable speed
c.	Helium pressure at circulator discharge, MPa/psia	0.097/14.0
d.	Circulator location	Bottom of reactor vessel
e.	Type	Two stage axial
f.	Helium shutoff valve	At circulator entrance

3.1.8 Shutdown Cooling Heat Exchanger

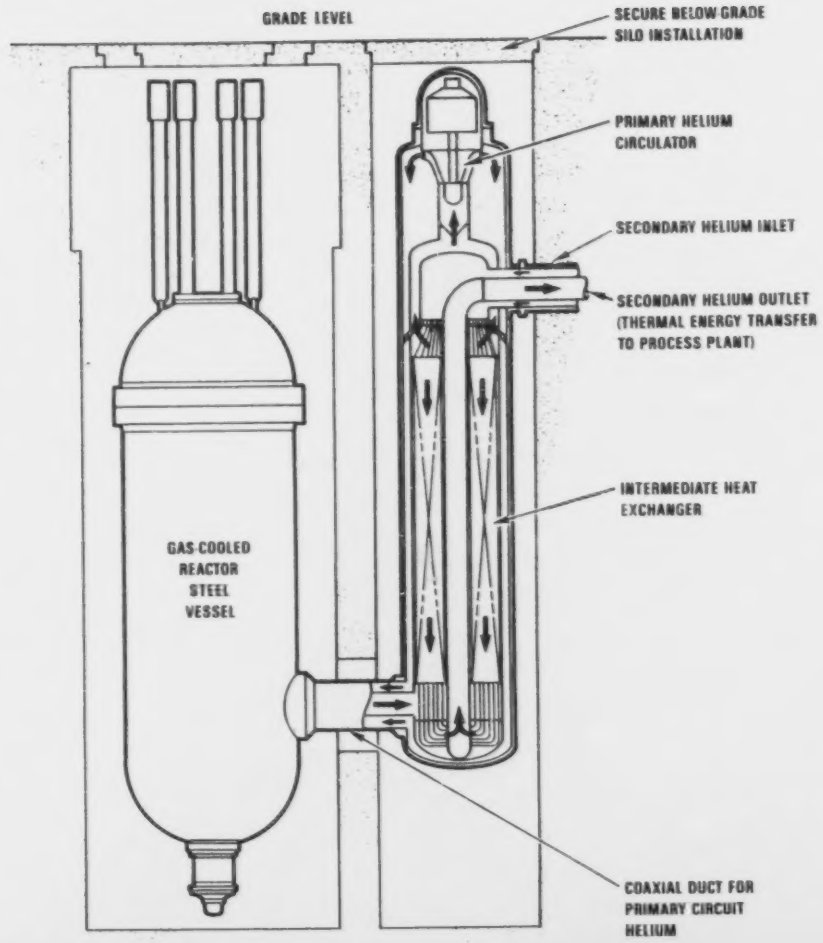
a.	Location	Reactor vessel
b.	Number of heat exchangers	1
c.	Heat duty, MW	3.3(6)
d.	Helium inlet temperature, °C/°F	246/474
e.	Helium outlet temperature, °C/°F	72/161
f.	Helium flow, kg/h/lb/h	13,300/29,320
g.	Cooling water inlet temperature, °C/°F	43/110
h.	Cooling water outlet temperature, °C/°F	71/159
i.	Cooling water pressure, MPa/psia	4.83/700
j.	Cooling water flow, kg/h/lb/h	105,240/232,000

(6) Depressurized semi-steady state condition.

ADVANCED HIGH EFFICIENCY MHTGR GAS TURBINE PLANT CONCEPT



ADVANCED MHTGR PLANT CONCEPT WITH INTERMEDIATE HEAT EXCHANGER FOR NUCLEAR PROCESS HEAT


GRADE LEVEL
SECURE BELOW GRADE SILO INSTALLATION
PRIMARY HELIUM CIRCULATOR
SECONDARY HELIUM INLET
SECONDARY HELIUM OUTLET (THERMAL ENERGY TRANSFER TO PROCESS PLANT)
INTERMEDIATE HEAT EXCHANGER
GAS-COOLED REACTOR STEEL VESSEL
COAXIAL DUCT FOR PRIMARY CIRCUIT HELIUM

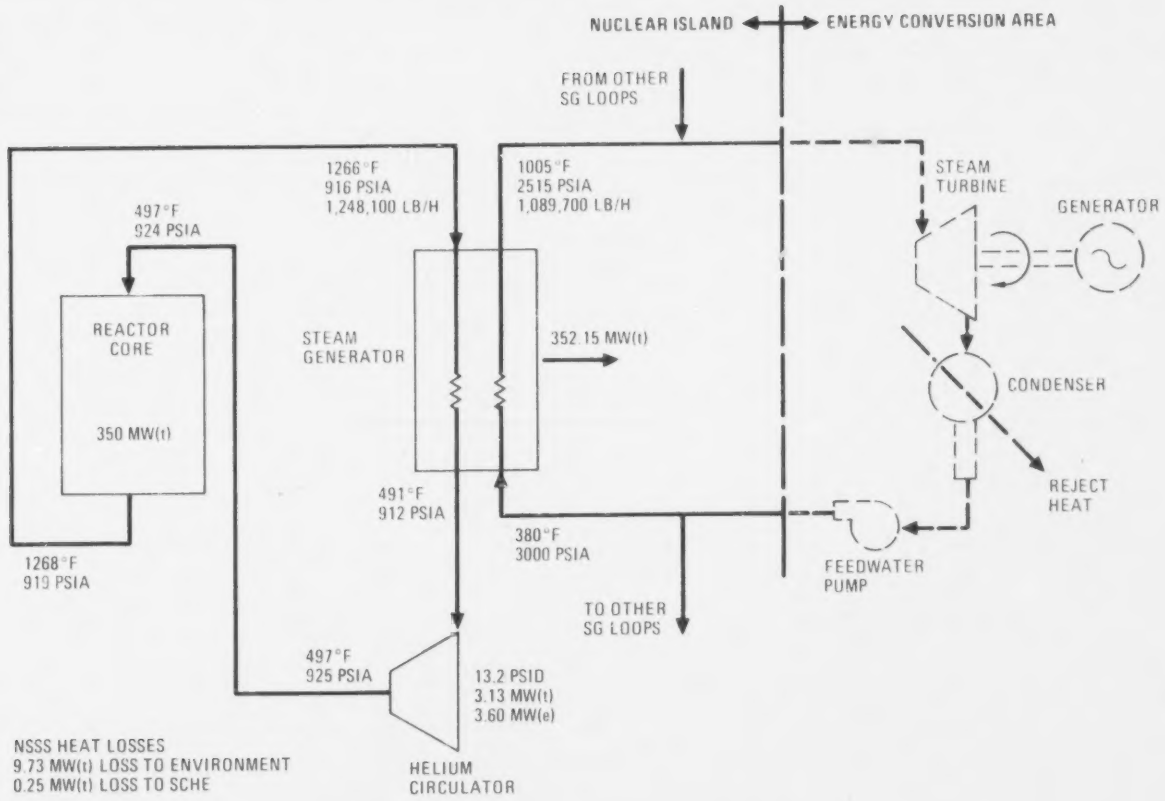
FUTURE MILESTONES

PROPOSED DOE MHTGR PROGRAM ACTIVITIES FOR FV-87

- COMPLETE CONCEPTUAL PLANT DESIGN
  - CONSTRUCTABILITY/FABRICABILITY ANALYSIS
  - DEFINITIVE COSTS AND SCHEDULE
  - COMPLETE TECHNOLOGY PLANS
- TEST OF REFERENCE FUEL IN TEST REACTOR AT ORNL
- INITIATE DEVELOPMENT TEST PROGRAMS ON KEY PROGRAMS, E.G., MAGNETIC BEARING CIRCULATOR
- OBTAIN PRELIMINARY SAFETY EVALUATION REPORT (PSER) AND LICENSABILITY STATEMENT FROM NRC
- ESTABLISH PROJECT STRATEGY

APPENDIX A

MODULAR HTGR CONCEPTUAL DESIGN DATA


NUCLEAR ISLAND
ENERGY CONVERSION AREA
FROM OTHER SG LOOPS
1266°F
916 PSIA
1,248,100 LB/H
1005°F
2515 PSIA
1,089,700 LB/H
STEAM TURBINE
GENERATOR
497°F
924 PSIA
REACTOR CORE
350 MW(t)
STEAM GENERATOR
352.15 MW(t)
CONDENSER
REJECT HEAT
491°F
912 PSIA
380°F
3000 PSIA
1268°F
919 PSIA
FEEDWATER PUMP
TO OTHER SG LOOPS
497°F
925 PSIA
13.2 PSID
3.13 MW(t)
3.60 MW(e)
NSSS HEAT LOSSES
9.73 MW(t) LOSS TO ENVIRONMENT
0.25 MW(t) LOSS TO SCHE
HELIUM CIRCULATOR

APPENDIX A (cont)

MHTGR - CONCEPTUAL DESIGN DATA

3.1.9	Decay Heat Removal Method	
	a. Pressurized heat removal method	Forced on SG or shutdown cooling heat exchanger or conduction/radiation to RCCS(7)
	b. Depressurized heat removal method	Forced on SG or shutdown cooling heat exchanger or conduction/radiation to RCCS(7)
3.2	REACTOR CAVITY COOLING SYSTEM	
	a. Number of loops	1
	b. Type	Air cooling panels
3.3	ULTIMATE HEAT SINK DESIGN	
	a. Type	Forced draft

(7) RCCS = Reactor Cavity Cooling System

Mrs. LLOYD. Thank you, Mr. Northup, and you certainly added to it. I appreciate your testimony.

Mr. Chernock, we know that your testimony will be equally valuable.

Mr. CHERNOCK. Thank you, Madam Chairman. It is always a pleasure to appear in front of this committee.

Thank you for this opportunity to testify on the progress being made on the modular high temperature gas reactor. My name is Warren Chernock, and I am vice president, Advanced Nuclear Systems, at Combustion Engineering. I have provided written testimony, and I request that it be made part of the record.

Mrs. LLOYD. Without objection.

Mr. CHERNOCK. This year the program participants have made notable progress in defining and documenting a specific design of a modular HTGR which will meet the goals and requirements that we carefully identified in previous years.

It was only a year ago that we selected an annular core configuration using prismatic fuel after completing a disciplined selection process. We have made significant progress developing the design details of this configuration.

All this has been done in an orderly, top-down fashion, so that we cannot only tell what the design is, but also how and why the design features were established.

Extensive documentation has been prepared largely for the purpose of enabling NRC to carry out a careful and detailed evaluation of the concept during the coming year. We are confident that this evaluation will enable them to issue a licensability statement early in fiscal year 1988 concurring that the MHTGR Program is on the correct path toward the design of a nuclear plant with inherent, passive safety.

The cornerstone of this documentation is the preliminary safety identification document. This is supported by a probabilistic risk assessment.

The fact that we have been able to prepare a meaningful PRA at this early stage is indicative of the amount of detailed work that has been performed, and also of the emphasis that we are putting on the systematic evaluation of safety-related requirements right from the start.

During fiscal year 1987 we expect to complete the conceptual design of the MHTGR to the point where we can furnish the committee with reliable, supportable answers to many of the questions which you have asked. We feel that it is more important to provide the right answers after careful study than to come forward early with estimates which may lead to disappointment later on.

One of the objectives of the year's work is, in fact, to develop the basis for a recommendation for a demonstration project.

Combustion Engineering believes that the licensing plan which was described in earlier testimony by DOE is realistic and will achieve its objectives. We will continue to make our current experience in working with NRC available to the program. In particular, we will make an important contribution in the area of PRA.

The key near-term activity will be the review of the PSID by NRC, leading to their issuing a licensability statement. We antici-

pate a number of meetings with them to discuss particular topics, their questions, and the preparation of amendments.

It is feasible to have a prototype MHTGR in operation in the mid-1990's, considering the state of the design and the technology today. Adequate funding would be needed to meet this schedule, as well as timely and resolute commitment to the program on the part of DOE, NRC, and the Congress.

I would like to commend this committee for its continued commitment to this program. Some of the key earlier activities would be: Project definition in fiscal year 1988; site selection during fiscal year 1988; project authorization for fiscal year 1989.

This schedule could lead to standard design certification by the NRC, following rulemaking, based on public hearings, by the end of fiscal year 1997. We look forward to commercial orders once a standard license has been obtained.

These documents are supported by the system and subsystem design descriptions covering the entire project. The remaining needs for development and testing have been specified in "Design Data Needs," prepared in a disciplined interaction between the designers and Oak Ridge.

The technology program will be based on and limited to those things which are identified as needed to support and validate the design. The program is thus rapidly narrowing in on a specific design and a highly focused technology program.

Combustion Engineering has participated actively in this process, both by contributing portions of the documents, and by reviewing the entire product. In particular, we have been able to bring to bear out nuclear expertise in the preparation of the PRA.

Our major design efforts have been on the vessels and heat exchangers. We have confirmed the feasibility of using steel vessels for this application, based on existing pressurized water reactor technology. We have confirmed the feasibility of factory fabrication and shipment to the site.

We have carried out seismic evaluations in conjunction with Bechtel, and developed a concept for a vessel support system compatible with seismic requirements and with the building structure. We have worked out with General Atomics a way to transmit the seismic loads to the core which meets the core requirements.

We can say that licensing basis events do not appear to challenge vessel integrity, based on the leak before break, an approach which has been accepted by the NRC.

We have confirmed the feasibility of the helical-coil steam generator and of the shutdown-cooling heat exchanger for the given design conditions. We have assessed the requirements for the given design conditions.

We have assessed the requirements for in-service inspection of these components, and means to meet them. We have identified and documented the design data needs for all these components.

Finally, we have prepared cost estimates and schedules.

During this year DOE has taken an important step toward integrating the management of the MHTGR Program by establishing the Plant Design Control Office. The office, located in La Jolla, CA, is staffed by permanent representatives seconded by each participant. It serves to review and coordinate all the activities on the

program to make sure that we are all working toward common objectives in an efficient manner.

We regard this as a most important step, and Combustion Engineering is pleased to have been able to supply the director of PDCO. This brings Combustion Engineering's extensive experience in NSSS project management and design integration into the service of this important program.

This office has already had an impact on the timely creation and review of the large amount of documentation which has been described previously.

In preparation for the work during fiscal year 1987, this office created a logic network detailing how these objectives were to be reached and every participant then prepared a subnetwork showing his proposed efforts in support of the common objectives. This has provided a logical framework base upon which DOE can make objective decisions of how to assign the available efforts.

As I pointed out previously, the technology program has been tied to the needs identified and documented by the designers. For those components which are C-E's particular responsibility, we have identified a number of data needs. None of these are feasibility, go-no-go matters; they involve confirmatory data which will be needed by the end of final design to justify and validate that design.

Most of them involve the ability of the equipment to withstand the design conditions for the full life the plant. In all cases, design alternatives have been identified in case the data are not forthcoming or do not meet assumptions. A summary of these R&D needs are included in my written testimony.

Test of a complete steam generator is not deemed necessary, based on the successful operation of similar steam generators at Fort St. Vrain and the THTR-3 in Germany. I would like to take this opportunity to point out to this committee that C-E has obtained at its own expense, a technical exchange and licensing agreement with Sulzer Bros., Ltd., of Switzerland, the suppliers of the steam generators for THTR in Germany.

This provides us with a substantial basis of design and operating experience in this specialized field. To meet the design goals which have been established for fiscal year 1987, that is the completion of the conceptual design to support a recommendation for a construction project, will require a minimum of $25 million.

We anticipate that substantially higher funding levels will be required in future years to meet the schedule suggested by the committee, but we would like to refrain from quoting specific numbers until the fiscal year 1987 cost and schedule estimates have been prepared.

We also strongly recommend that appropriate funding be provided to NRC for PSID review and other MHTGR activities requiring NRC interaction.

As you will note in my testimony before this committee on February 18, 1985, I was somewhat critical of coordinated progress on the HTGR effort. I am now pleased to comment most favorably on the progress that has been made since then and the impact of the orderly top-down approach in defining and documenting a specific

MHTGR reference design which all program participants are now focused upon.

This program has leadership and momentum. DOE is to be commended for its outstanding performance and we urge you, Madam Chairman, and Members of Congress, to continue this momentum via adequate funding in fiscal year 1987, and in future years.

Thank you for the opportunity to present our views on the status of the MHTGR to your committee and to provide Combustion's continuing support for the success of this program.

[The prepared statement of Mr. Chernock follows:]

STATEMENT SUBMITTED TO

SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION

COMMITTEE ON SCIENCE AND TECHNOLOGY

UNITED STATES HOUSE OF REPRESENTATIVES

MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR PROGRAM REVIEW

Warren P. Chernock
Vice-President, Advanced Nuclear Systems
Nuclear Power Systems
Combustion Engineering, Inc.
September 17, 1986

Madam Chairman and Members of the Committee, thank you for this oppportunity to testify on behalf of Combustion Engineering on the progress being made on the Modular High Temperature Gas Reactor. My name is Warren P. Chernock, and I am Vice-President, Advanced Nuclear Systems, at C-E.

PROGRAM ACCOMPLISHMENTS IN FY-1986

This year the program participants have made notable progress in defining and documenting a specific design of a modular HTGR which will meet the goals and requirements which have been set out for it. You will recall that in previous years we carefully identified these goals and requirements, trying to make sure that they were complete, accurate, and clearly stated. We then examined in a systematic way the various configurations which might be used to meet these goals. A year ago we selected an annular core configuration using prismatic fuel, and documented the reasons for this choice. Now we have developed a more detailed design of this configuration. All this has been done in an orderly, top-down fashion, so that we cannot only tell what the design is, but also how and why the design features were established.

Extensive documentation has been prepared this year, largely for the purpose of enabling NRC to carry out a careful and detailed evaluation of the concept during the coming year. We are confident that this evaluation will enable

them to issue a Licensability Statement early in FY-1988 concurring that the MHTGR Program is on the correct path toward the design of a nuclear plant with inherent, passive safety.

Madam Chairman, as you well know, one of the main reasons for highly publicized cost overruns of nuclear projects has been that the regulatory requirements which had to be met before operation could begin were not identified at the start, when the initial cost estimates were made. We are trying very hard to avoid this problem by taking the time now to work out exactly what these requirements should be, and to obtain NRC concurrence. This is time well spent, and we are confident that it will avoid much grief later on.

The cornerstone of this documentation is the Preliminary Safety Identification Document, or PSID. This is supported by a Probabilistic Risk Assessment (PRA). The fact that we have been able to prepare a meaningful PRA at this early stage is indicative of the amount of detailed work that has been performed, and also of the emphasis that we are putting on the systematic evaluation of safety-related requirements right from the start.

DESIGN ACTIVITIES IN FY-1987

During FY-1987 we expect to complete the conceptual design of the MHTGR to the point where we can furnish the Committee with reliable, supportable answers to many of the

questions which you have asked. We feel that it is more important to provide the right answers after careful study than to come forward early with estimates which may lead to disappointment later on. One of the objectives of the year's work is, in fact, to develop the basis for a recommendation for a construction project.

High points of C-E's planned design activities include the following:

- Design of the means for supporting the steam generator tube bundle to withstand seismic loads
- Design to accommodate thermal transients, especially in the superheater and the hot tube sheet
- Validation of the thermal-hydraulic codes TEMPEST and COMMIX for the three-dimensional geometry of the steam generator
- Determine the feasibility of and design the sliding seals between hot and cold helium in the steam generator
- Size and analyze the design for the specific system parameters which will be selected early in the fiscal year
- Develop in greater detail the means to meet requirements for in-service inspection
- Prepare updated cost and schedule estimates.

LICENSING

C-E believes that the Licensing Plan which has been

described in earlier testimony by DOE is realistic and will achieve its objectives. We will continue to make our current experience in working with NRC available to the program. In particular, we think we can make an important contribution in the area of PRA. The key near-term activity will be the review of the PSID by NRC, leading to their issuing a Licensability Statement. We anticipate a number of structured meetings with them to discuss particular topics, less formal meetings to discuss their questions, and the preparation of amendments.

PROTOTYPE OPERATION IN THE MID-90's

It is feasible to have a prototype MHTGR in operation in the mid-90's, considering the state of the design and the technology. Adequate funding would be needed to meet this schedule, as well as timely and resolute commitment to the program on the part of DOE, NRC, and the Congress. Some of the key early activities would be:

- Project definition in FY-87
- Site selection during FY-88
- Project authorization for FY-89

This schedule could lead to Standard Design Certification by the NRC, following rulemaking based on public hearings, by the end of FY-97. We look forward to commercial orders once a standard license has been obtained.

These documents are supported by System and Subsystem Design Descriptions covering the entire plant. The remaining

needs for development and testing have been specified in "Design Data Needs" prepared by the designers. The technology program will be based on, and limited to, those things which are identified as needed to support and validate the design. The program is thus rapidly narrowing in on a specific design and a highly focused technology program.

C-E ACTIVITIES

C-E has participated actively in this process, both by contributing portions of the documents, and by reviewing the entire product. In particular, we have been able to bring to bear our nuclear expertise in the preparation of the PRA.

Our major design efforts have been on the vessels and heat exchangers. We have confirmed the feasibility of using steel vessels for this application, based on accepted PWR technology. We have confirmed the feasibility of factory fabrication and shipment to the site. We have carried out seismic evaluations in conjunction with Bechtel, and developed a concept for a vessel support system compatible with seismic requirements and with the building structure. We have worked out with GA a way to transmit the seismic loads to the core which meets core requirements. We can say that licensing basis events do not appear to challenge vessel integrity, based on "leak-before-break," an approach which has been accepted by the NRC.

We have confirmed the feasibility of the helical-coil steam generator and of the shutdown-cooling heat exchanger for the given design conditions. We have assessed the requirements for in-service inspection of these components, and means to meet them. We have identified and documented the Design Data Needs for all these components. Finally, we have prepared cost estimates and schedules.

MANAGEMENT ACCOMPLISHMENTS

During this year DOE has taken an important step toward integrating the management of the MHTGR program by establishing the Plant Design Control Office. The PDCO, located in La Jolla, California, is staffed by permanent representatives seconded by each participant. It serves to review and coordinate all the activities on the program to make sure that we are all working toward a common objectives in an efficient manner. We regard this as a most important step, and C-E is pleased to have been able to supply the Director of PDCO. This brings C-E's extensive experience in NSSS project management and design integration into the service of this program.

PDCO has already had an impact on the timely creation and review of the large amount of documentation which has been described previously. In preparation for the work during FY-1987, PDCO defined the objectives for the year, and then created a logic network detailing how these objectives were to be reached. Every participant then prepared a sub-network

showing his proposed efforts in support of the common objectives. This has provided a logical framework base upon which DOE can made objective decisions of how to assign the available funding.

REMAINING R&D NEEDS

As I pointed out previously, the technology program has been tied to the needs identified and documented by the designers. For those components which are C-E's particular responsibility, we have identified a number of data needs. None of these are feasibility, "go-no-go" matters; they involve confirmatory data which will be needed by the end of final design to justify and validate that design. Most of them involve the ability of the equipment to withstand the design conditions for the full life of the plant. In all cases, design alternatives have been identified in case the data are not forthcoming or do not meet assumptions.

Radiation embrittlement of the vessel falls outside the design envelope of the PWR because the temperature is somewhat lower, thus reducing the annealing of effects of radiation. However, the fluence is substantially lower. Nevertheless, confirmatory data will be needed.

In the event of a highly unlikely loss-of-coolant event, parts of the vessel are estimated to reach a temperature in excess of 800°F as the decay heat is safely dissipated to the reactor cavity by radiation and conduction. This

is in excess of the values incorporated in the ASME codes, although material property data exist and support our conclusion that the unpressurized vessel will not be damaged.

Materials property data are needed for the steam generator materials for the design service conditions. The emphasis is on long-term creep and fatigue properties in a dry helium atmosphere. The materials involved are 2 ¼ Cr - 1 Mo, alloy 800H, and the bimetallic weld between them.

To support the steam generator design, special effects tests are needed, including:

- Accoustic and vibration tests
- Flow distribution tests
- Tests of vibrational fretting and sliding wear
- Tests of tube wear-protection devices.

No test of a complete steam generator is deemed necessary, based on the successful operation of similar steam generators at Ft. St. Vrain and the THTR-300 in Germany. I would like to take this opportunity to point out to this Committee that C-E has obtained, at its own expense, a technical exchange and license agreement with Sulzer Brothers, Ltd. of Switzerland, the suppliers of the steam generators for the THTR. This provides us with a substantial basis of design and operating experience in this specialized field.

FUNDING REQUIREMENTS

To meet the design goals which have been established

for FY-87, that is, the completion of the Conceptual Design to support a recommendation for a construction project, will require a minimum of $25 million. We anticipate that substantially higher funding levels will be required in future years to meet the schedule suggested by the Committee, but we would like to refrain from quoting specific numbers until the FY-87 cost and schedule estimates have been prepared.

At this funding level, only the most essential activities can be carried out. This is little contingency to take care of adverse findings. In particular, we at C-E cannot make as large a contribution to the program as we would like to.

SUMMARY

As you will note in my testimony before this Committee on February 18, 1985, I was somewhat critical of coordinated progress on the HTGR effort. I am now pleased to comment most favorably on the considerable progress which has been made since then and the impact of the orderly top-down approach in defining and documenting a specific MHTGR reference design which all program participants are now focused upon. This program has leadership and momentum. DOE is to be commended for its outstanding performance and we urge you, Madam Chairman, and Members of this Committee, to continue this momentum via adequate funding in FY-1987 and in future years.

Thank you for the opportunity to present our views on the status of the MHTGR to your Committee.

Mrs. LLOYD. Thank you very much, Mr. Chernock, for your testimony.

Dr. Kasten if you will proceed?

Mr. KASTEN. Thank you, Madam Chairman, Members of Congress. It is a pleasure to appear before this subcommittee; and thank you for the privilege of presenting testimony today.

Oak Ridge National Laboratory is supportive of the modular high-temperature gas cooled reactor, and serves as the lead laboratory for base technology development. The testimony provided at this time pertains to the status and requirements of base technology development.

Base technology development—before I proceed, I would like to show some vu-graphs, if we may have a vu-graph machine.

Mrs. LLOYD. We welcome your vu-graphs.

[Slide show.]

Mr. KASTEN. Base technology development provides MHTGR design and licensing activities with need fundamental information about the properties and behavior of components and systems. Present technology is sufficient to show the feasibility of the MHGTR concept and to develop the conceptual design.

Base technology programs still needed have been identified and focused on the specific requirements of the concept under development. The technology programs described here, develop the basis correlations concerning the behavior of physical and chemical phenomena as needed for the design of components and systems.

These programs also provide experimental validation of mathematical models predicting system behavior necessary for licensing. For example, retention of fission products.

The specific development areas discussed here are: fuels performance and fission product behavior; graphite behavior; and metals behavior.

First fuel and fission products technology. Fission product retention within the fuel coatings is the key ingredient to the success of achieving passive safety in the modular HTGR. And I will emphasis this.

This figure shows that a triso-coated fuel particle. The fuel kernel is a mixture of UO–2 and UC–2, termed uranium oxycarbide. It is surrounded by a porous layer, or buffer layer, of pyrolytic carbon which provides expansion volume. Surrounding the buffer layer are successive layers of pyrolytic carbon, silicon carbide, and pyrolytic carbon, which together contain the fission products.

These coated particles have high fission product retention capability, even at high temperatures, as show in the next vu-graph.

It is now well established that the high-quality coated fuel particles retain essentially all fission products under postulated accident conditions as well as normal operating conditions.

This strongly supports the feasibility of the MHTGR concept. However, coated fuel particles have to be fabricated into rods and assembled in the fuel elements—as show in the next vu-graph.

These steps involve fabrication processes which can lead to some breakage of fuel coatings. The Fuel Technology Development Program concentrates on economic processes which maintain the integrity of the fuel coating and on performance testing of the fuel

rods that are fabricated under both normal reactor as well as postulated accident conditions.

A small amount of uranium contamination will be present in the fuel rod matrix material as a result of the coating process. Further, a small fraction of coating defects are expected to occur during the coating process.

As a result, fission product behavior throughout the reactor system needs to be known on a quantitative basis in order to meet the no-evacuation planning of the program. Fission products released from the fuel rod can be transported from the core—and the next vu-graph shows the reactor system.

The reactor core is in the center of the larger vessel, and the fission products can be transported from the core to the steam generator area, which is shown the smaller vessel, and this plateout will tend to build up over time. During rapid depressurization, some of these fission products could be released through the atmosphere.

Present knowledge indicates there is sufficient retention of fission products during rapid depressurization, but more quantitative information is required. A definitive program has been developed for demonstrating specific behavior and for validating the mathematical models used to predict fission product transport.

The total cost of the Fuel and Fission Product Technology Development Program is estimated to be about $36 million over 7 years.

Concerning graphite technology; graphite is an important material in the reactor and constitutes a major volume fraction of the core region—as shown in the figure to the left.

The next figure shows a better view of the annular characteristics of the core and the inner and outer reflector graphite regions. A shaded area represents the fueled region and the other areas represent the reflector regions.

Great Lakes Research Corp. grade H-451 graphite has been chosen for the fuel elements and the replaceable reflector elements. Stackpole Corp. grade 2020 graphite has been chosen for the permanent side reflectors and the core support structures.

There is much information on graphite behavior in gas-cooled reactors in general supporting the feasibility concept that we are developing. At the same time, additional physical and chemical property data is needed for the specific MHTGR graphite to support the detailed design of components under normal operating conditions and under postulated accident conditions.

The Graphite Technology Development Program will obtain the needed data base which involves evaluating variations in graphite strength within and between billets from which the elements are made, involves determining radiation creep, which is an important stress reliever, and involves reducing oxidation characteristics of the graphite.

Most of the Graphite Technology Program is related to obtaining physical property data required for reliable component design and reducing associated investment risk. The cost of the Graphite Program is estimated to be about $11 million over 6 years.

With regard to metals technology, metals technology for the MHGTR is sufficiently known to assure the feasibility of metal components. Additional information relates primarily to property

data bases required for reliable component design and reducing associated investment risk.

A Metals Technology Development Program has been identified which will provide the required material property correlations and addresses the technology requirements of the metallic components of the MHTGR.

The key components are shown in the next vu-graph, and consist of the reactor vessel, which is a light-water, reactor-type vessel steel; and the steam generator vessel, the cross duct which contains hot gas duct; the core barrel system, which is inside the reactor vessel and outside of the reactor core; and the steam generator, in the steam generator vessel.

Long-term testing is needed to obtain the materials property data base, with experimental studies of fracture mechanical properties, mechanical properties, radiation effects, and environmental and corrosion effects, and weld properties. An initial of the Metals Technology Development Program cost is $10 million over 5 years.

With regard to overall cost estimates—may I have the next vu-graph, please—the overall estimate for the base Technology Program is about $57 million, as shown here. This cost and cost breakdown can be compared with previous cost estimates given in the next table.

These previous estimates are based on the conventional HTGR designs of the past. As shown, the estimated cost for fuel and associated process development has decreased substantially from the early 1980's.

The decreased cost are due largely to a significant progress in developing high-performance fuel and to better understanding process development requirements. The costs for determining fission product behavior have increased, primarily because in the MHTGR we want those fission products to stay within the system, and in previous times we were counting on containment to retain fission products rather than the reactor system.

In graphite and metals development cost estimates have decreased substantially. The decrease in graphite development is due primarily to the reduced safety concerns relative to failure of graphite components in the MHTGR. In the metals area, the development program has decreased due to fewer and less complicated components in the MHGTR.

With regard to international cooperation and base technology, the United States and the Federal Republic of Germany presently carry out a cooperation under an umbrella agreement involving MHTGR technology development.

Also, the United States and Japan are cooperating in MHTGR research and development. The United States/Federal Republic of Germany cooperation involves a subprogram in metals development, in fuels and fission products, and graphite behavior.

The United States/Japan cooperation involves fuel testing, including testing of fuels at high temperatures and it is planned to have a Graphite Cooperative Program and a metals cooperative as proposed.

These cooperative programs provide base technology information pertinent to the MHGTR, and continuing and expanding such coop-

eration can have a beneficial effect on increasing the physical data base and limiting overall development costs.

In summary, base technology development for the modular high temperature gas-cooled reactor, proves the basic information on which the design and licensing of the system proceeds in particular on material properties, on fission product retention capabilities. Present information is sufficient to show the feasibility of the concept and to develop the conceptual design.

The Technology Development Programs still needed have been identified, and are based on the specific needs of the MHTGR concept. These programs will develop the material data bases and validate the system response models which are used in the detailed design and licensing process.

The planned base technology program is relative modest. It is well focused on the needs. Continued emphasis on international cooperation in base technology development would increase the amount of physical property in the data base as well as limit development costs.

Thank you.

[The prepared statement of Mr. Kasten follows:]

OVERVIEW OF MHTGR BASE TECHNOLOGY

STATEMENT BY

PAUL R. KASTEN
Technical Director for
Gas-Cooled Reactor Program
Oak Ridge National Laboratory

before the
SUBCOMMITTEE ON ENERGY RESEARCH
AND PRODUCTION
of the
COMMITTEE ON SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES

September 17, 1986

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
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STATEMENT ON
OVERVIEW OF MHTGR BASE TECHNOLOGY
DR. PAUL R. KASTEN, TECHNICAL DIRECTOR
GAS-COOLED REACTOR PROGRAM
OAK RIDGE NATIONAL LABORATORY

BEFORE THE SUBCOMMITTEE ON
ENERGY RESEARCH AND PRODUCTION

U.S. HOUSE OF REPRESENTATIVES
COMMITTEE ON SCIENCE AND TECHNOLOGY

September 17, 1986

Madam Chairman and Members of the Subcommittee:

Thank you for the privilege of presenting testimony today. Oak Ridge National Laboratory is supportive of the Modular High-Temperature Gas-Cooled Reactor concept and serves as the lead laboratory for base technology development in the MHTGR program. Testimony provided at this time pertains to the status and requirements of base technology development for the MHTGR.

Base technology development provides MHTGR design and licensing activities with needed basic information on material properties and behavior. At this time, existing technology is sufficient to show the feasibility of the MHTGR concept and to develop the conceptual design. Further, the base technology development programs still needed have been identified and focus on the specific MHTGR requirements.

The base technology programs develop the basic correlations concerning the behavior of physical and chemical phenomena; they also provide experimental validation of mathematical models predicting system behavior necessary for licensing (e.g., retention of fission products). The specific development areas discussed here are (1) fuels performance and fission product transport, (2) graphite behavior, and (3) metals behavior.

Fuel and Fission Products Technology. Fission product retention within the fuel coatings is the key ingredient to achieving passive safety in the MHTGR, and this area will be emphasized. Figure 1 illustrates a Triso-coated fuel particle used in the MHTGR. The fuel kernel is a mixture of UO_2 and UC_2, termed uranium oxycarbide; it is surrounded by a porous layer (or buffer layer) of pyrolytic carbon which provides expansion volume. Surrounding the buffer volume are successive layers of pyrolytic carbon, silicon carbide, and pyrolytic carbon, which together contain the fission products. These coated fuel particles have very high fission product retention capability, even at high temperatures. It is now well established that high-quality coated fuel particles retain essentially all

fission products under anticipated reactor temperature conditions, as indicated in Figure 2; this strongly supports the feasibility of the MHTGR concept. However, coated fuel particles have to be fabricated into fuel rods and assembled into fuel elements, as illustrated in Figure 3. This involves fabrication processes which can lead to some breakage of fuel particle coatings. The fuel technology development program concentrates on (1) developing economic processes which maintain the integrity of the fuel particle coating, and (2) performance testing of fuel rods under normal reactor conditions as well as under postulated accident conditions.

A small amount of uranium contamination will be present in the fuel rod matrix material as a result of the coating process. Further, a small fraction of coating defects are expected to occur during the coating process. As a result, fission product behavior within the reactor system needs to be quantified in order to assure meeting the user requirements of "no-evacuation planning." Fission products released from the fuel rod can be transported to the heat exchanger system where "plateout" tends to occur on the cooler surfaces. Figure 4 shows the relative location of the steam generator and the reactor core. Over a period of time, fission products will "build up" on the steam generator surfaces. During rapid depressurization, some of these fission products could be released to the atmosphere. Present knowledge indicates there is sufficient retention of fission products during rapid depressurization, but more quantitative knowledge is needed. A definitive program has been developed for demonstrating specific behavior and for validating the mathematical models used to predict fission product transport behavior.

The total cost of the fuel and fission product technology development program is estimated to be about $36M over seven years.

Graphite Technology. Graphite is an important material in the reactor and constitutes a major volume fraction of the core region. Figure 4 shows the general location of the fueled core, while Figure 5 illustrates more clearly the annular characteristics of the core and the inner and outer reflector graphite regions. Great Lakes Research Corporation grade H451 graphite has been chosen for the fuel elements and the replaceable reflector elements. Stackpole Corporation grade 2020 graphite has been chosen for the permanent side reflectors and the core support structures. There is much information on graphite behavior in gas-cooled reactors which generally supports the feasibility of the MHTGR concept. At the same time, additional physical and chemical property data is needed to support the detailed design of graphite components under normal operation conditions and under postulated accidents. The graphite technology development program will obtain the needed data base and involves evaluating variations in graphite strength determining irradiation creep behavior (an important mechanism for relieving stresses), and reducing oxidation characteristics. Most of the graphite technology program effort is related to obtaining property data required for reliable component design and reducing associated investment risk. The cost of the graphite program is estimated to be about $11M over six years.

Metals Technology. The metals technology for the MHTGR is sufficiently known to assure the feasibility of metal components. Additional information relates primarily to property data bases required for reliable component design and reducing associated investment risk. A metals technology development program has been identified which will provide the required material property correlations. The program addresses the technology requirements of the metallic components of the MHTGR; the pertinent components are included in Figure 4 and consist of the reactor vessel (constructed of SA 533B steel, which is a Light-Water Reactor pressure-vessel steel); the duct which connects the two vessels (made of Alloy 800H); the core lateral restraint system (includes the core barrel) surrounding the graphite core (made of 2.25Cr-1Mo and Alloy 800H); and the steam generator (made of Alloy 800H and 2.25Cr-1Mo steel). Long-term testing is needed to complete the materials property data base; key measurements involve fracture mechanics, mechanical properties, radiation effects, environmental and corrosion effects, and weld properties. An initial estimate of the metals technology development program cost is about $10M over five years.

Cost Estimates. The overall estimate for the base technology program is about $57M, as shown in Table 1. This cost and cost breakdown can be compared with previous estimates given in Table 2 (the previous estimates were for conventional HTGRs.) As shown, the estimated cost for fuel and associated process development has decreased substantially from 1980. The decreased costs are largely due to significant progress in developing high-performance fuels and better understanding of process development requirements. The costs for determining fission product behavior have increased, primarily because the MHTGR places emphasis on retaining essentially all fission products within the reactor system. The graphite development cost estimate decreased substantially from 1980 to 1986, as has the metals estimated cost. The decrease in graphite development is due primarily to the reduced safety concerns relative to failure of graphite components in MHTGRs. In the metals area, previous estimates included R&D for ceramics associated with the core support posts and with insulation components not present in the MHTGR. As a result, the metals development program has decreased due to fewer and less complicated components in the MHTGR.

International Cooperation and Base Technology. The United States and the Federal Republic of Germany presently carry out a cooperation under an Umbrella Agreement covering MHTGR technology development. Also, the United States and Japan are cooperating in MHTGR research and development. The U.S./FRG cooperation involves (1) a subprogram in metals development which addresses eight specific studies and (2) a subprogram in fuels, fission products, and graphite which covers sixteen specific investigations. The U.S./Japan cooperation involves testing of MHTGR fuels including their performance at high temperatures; further cooperation is planned in the graphite area and is being proposed in the metals area. These cooperative agreements provide the U.S. with base technology information pertinent to

the MHTGR. Continuing and expanding such cooperation can have a beneficial effect on increasing the amount of property-data information and limiting development costs.

Conclusions. Base Technology Development for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) provides basic information on material properties and on fission product retention capabilities. Present information is sufficient to show the feasibility of the MHTGR concept, and to develop the conceptual design. The base technology development programs required have been identified based on the specific needs of the MHTGR concept; these programs will develop the material data bases and validate the system response models required for detailed design and for licensing. The planned base technology program is relatively modest, being estimated at about $58M over seven years, and is well focused on the needs of the MHTGR concept. Continued emphasis on international cooperation in base technology development would increase the amount of property behavior information as well as limit development costs.

FIGURE 1

TRISO COATED FUEL PARTICLES RETAIN FISSION PRODUCTS AT HIGH TEMPERATURES

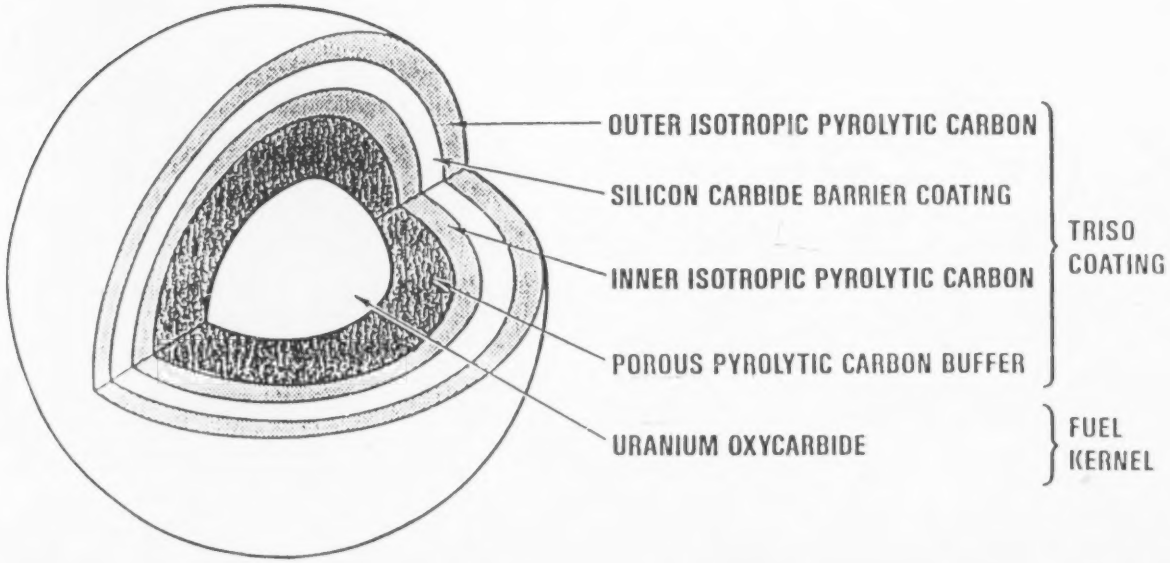
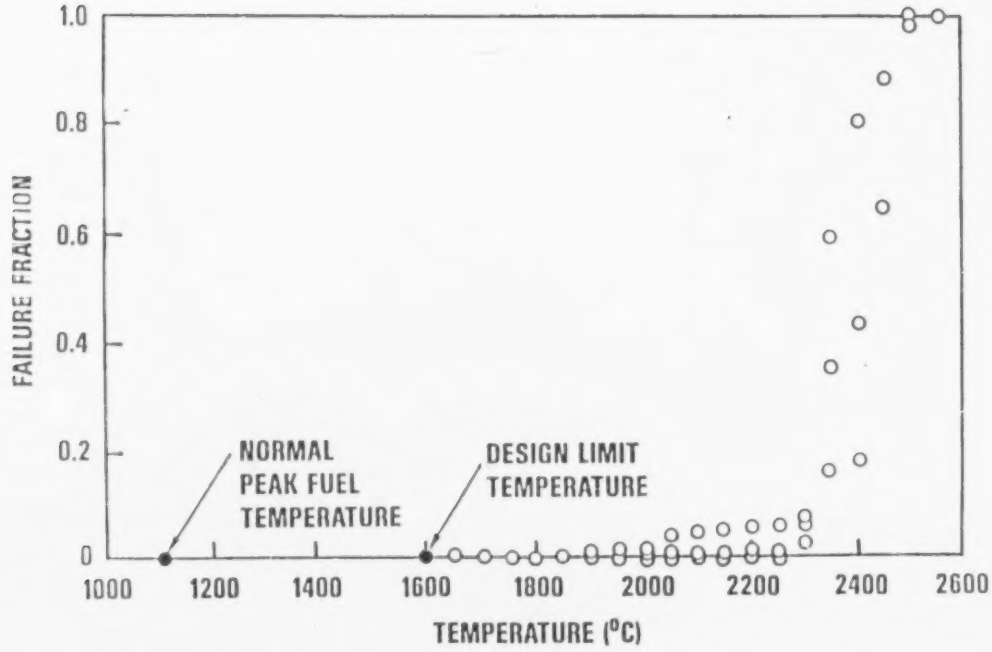


FIGURE 2

COATED FUEL PARTICLES MAINTAIN INTEGRITY AT HIGH TEMPERATURES



Mrs. LLOYD. Thank you, Paul, I think you are a good example of why it is a source of pride to me to represent the Oak Ridge community, but certainly the professional people at Oak Ridge give of their lives to enhance the potential of all mankind. We are very proud of what you and your associates are doing in regard to the MHTGR.

Thank you for being here.

Mr. KASTEN. Thank you.

Mrs. LLOYD. Dr. Kendall, we are awfully happy to have you here also.

Please proceed.

Mr. KENDALL. Thank you, Madam Chairman. I am Jim Kendall, I am director of program development for GCRA.

I would like to say a couple of things about GCRA from my perspective. I initially got involved as a member of the management committee of GCRA, representing Yankee Atomic Electric Co., and one of the things beyond the general merits of the HGTR technology that drew me further into this activity was the great responsiveness of the GCRA staff to the utility representatives, and the active involvement of the utility members in the process.

To my mind, when I use the term GCRA, I mean not just the GCRA staff but the utility membership because there is several layers of working groups, committees, that direct and support GCRA. So, with that background I'll go through primarily the scope of utility initiative activities that have developed over about the past 18 months.

We do have some involvement in the DOE Program, but I think that has been adequately addressed elsewhere.

You have seen some of the material on the project initiative through the joint statement, which had some brief references to it, then a more detailed reference in a submittal to the Secretary of Energy, back in July. So, just going through it briefly, I was also in addition to being on the management committee, chairman of a project strategy subcommittee that basically laid the groundwork for the utility initiative that we have brought to the point of the project strategy plan.

Just going briefly through it, the fundamental activity has been a project definition study that was funded by GCRA and that study looked at if we were to set out to do a demonstration plant, about what would you want do with it, how much would it cost, what kind of a schedule would you propose, and to try to put some kind of a framework in it.

The basic approach as to say initially that funding and constrained program that was well focused and then come back and look at what resources we have. I won't really go through the results of that study.

A couple of points of interest to me. One being that the field direct cost estimates and admittedly this is a very preliminary estimate, but with a very detailed code of accounts, amount to about $1,700 a kilowatt, which from a utility standpoint for a first-of-a-kind singular modular plant, is very promising.

On the other hand, the indirect cost for engineering alone amounted to about $2,000 per kilowatt, which is somewhat intimidating. I think it probably reflects where we are in terms of the

high cost of engineering and licensing analysis in the environment we are in.

I would point out from one of my perspectives, is the basic model draws heavily from the Yankee model, but on a more national basis than regional basis. And in the case of Yankee an escalated design cost would amount to about $25 million for a plant of almost identical size to what we are talking about here, for a total of around $300 million. So it is a tremendous difference.

We are working to try to get that down if we can, but I suppose that in large part it reflects two factors really. One, the legitimate factor of trying to look at assuring that this plant, in fact, can be replicated as a commercial plant.

Second one being the environment that we work in. But that won't stop us from trying to look hard at where we can get these costs down.

As far as the role—one thing I should make clear is that the strategy plan itself is, in fact, representation only of GCRA. We have informed and had participation of the other program participants but at this point our purpose was to establish an initial framework to go forward with the other program participants to see if we can identify a unified way to move forward and representing an initial view from the utility standpoint.

The role of the utilities that are proposed in the strategy plan, as I said, in my own mind it basically reflects a similar role to what Yankee Atomic played in the course of that development; I think there are a lot of parallels that could be drawn there.

From my time at Yankee I took several weeks of lunch hours and evenings and weekends going back through the documentation of the Yankee project, reading all the transmittals back and forth, and contacts and so forth, to get a better feel for how that worked.

I think that there is some real benefits, not only in the demonstration project itself, but commercialization to follow, of having an involved utility owner entity directly involved in the demonstration plant. We are not interested in designing the plant, but we are interested in being involved as day-to-day participant in the management of that demonstration project.

So, one of the things that goes with that is the question of coming up with significant funding, I don't think you can anticipate that kind of a role without significant funding. The project initiative, as indicated in the strategy plan, does envision a major funding portion from the utilities side.

That is one of the big question marks that goes with the plan, and I certainly can't say to you today that those funds, in fact, will be forthcoming. We have a long way to go before we are there.

But I would say that the plan envisions a step-by-step process, and starting now with yearly commitments and basically moving on to a multiyear commitment as we get several issues more defined. The one positive thing that I can say is that the GCRA board of directors, meeting about a month ago, authorized a target GCRA budget for 1987 that is consistent with that plan.

We intend to do our part to try to move that forward. It certainly is not an easy thing to do, and especially in light of the Chernobyl events which has had major effects on the ability of any utility to consider identification with any kind of a nuclear initiative.

It certainly makes it much more difficult right now. I think that may be getting better as time goes on. But we will continue to work that part of the problem.

We hope to be moving forward with the basic framework that we have outlined and working with the other participants to identify just how we do integrate all of these viewpoints and how we move forward.

I would like to just make an observation though, from my own perspective, and I think the perspective that I have gained from working as the chairman of the project strategy subcommittee, and working with the other utility people there.

In terms, as we understand, the DOE basic planning to move forward is moving toward a selection of a concept of technology and perhaps moving toward a demonstration plant with the intent that it would be an industry-led project with DOE in a supporting role. To my mind, the key factors that will make that possible would be that, No. 1, that obviously the user or the participants in the industry that are going to provide those funds and that involvement, have to be deeply involved in both the establishment of the selection criteria, the selection process itself, and have to begin to take an active role in the development of the technology.

We have to start a step at a time. We don't have a commitment of $400 million from utilities to date. We do have a past expenditure of approximately $4 million for 1986. We are approaching $2 million investment in the project strategy plan. I think that in itself is a strong indication that there is really utility interest there.

We are all working to see if we can find a way to bring this whole thing together and obviously there are many issues that swirl around it.

I would like to close by saying that one thing is certain that the support of this subcommittee has been a major factor for us in terms of talking to the utilities and in general, in bringing our initiative this far. We have a lot of work to do.

We will try to continue to build on that support. And I hope that we can come back next spring to you, Madam Chairman, and tell you that we are moving forward.

Thank you.

[The prepared statement of Mr. Kendall follows:]

FIGURE 3

MHTGR FUEL COMPONENTS

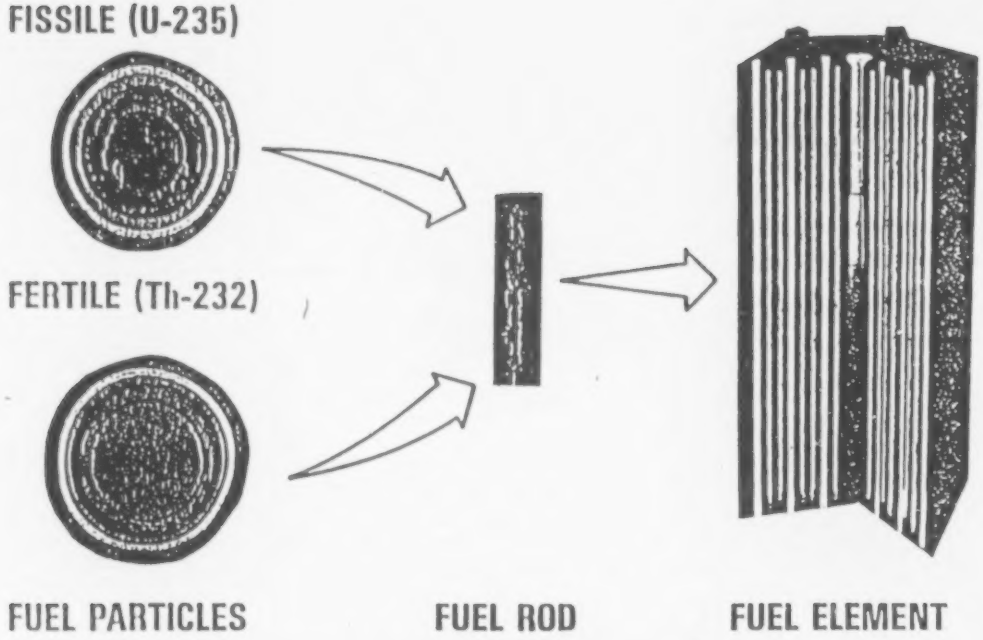


FIGURE 4

350 MW(t)
MODULAR
HTGR
ELEVATION

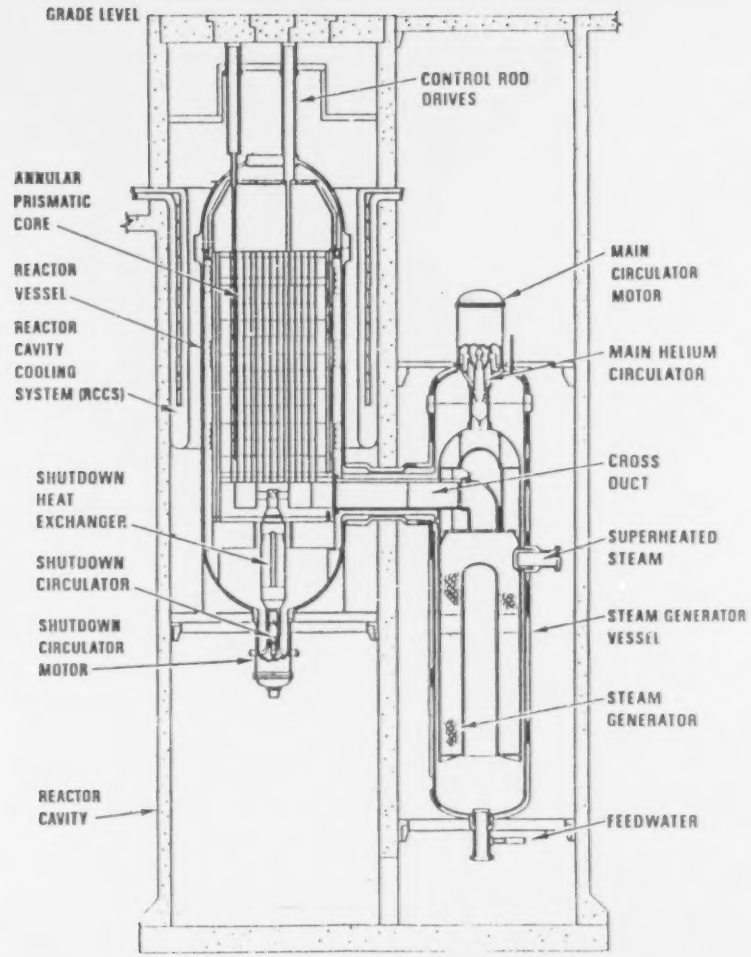


FIGURE 5

350 MW(t) MODULAR
REACTOR CORE CROSS SECTION

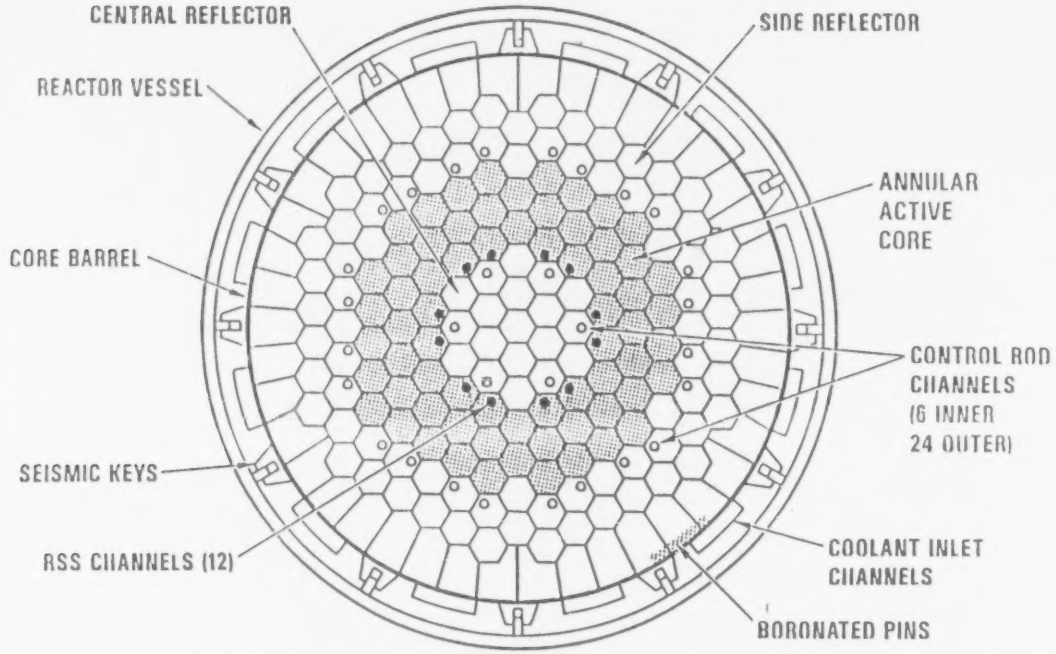


TABLE 1

INITIAL ESTIMATE OF BASE TECHNOLOGY PROGRAM COSTS	
FUELS AND FISSION PRODUCTS	$35,700K
GRAPHITE	$10,900K
METALS	$10,100K
TOTAL	$56,700K

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TABLE 2

COMPARISON OF BASE TECHNOLOGY DEVELOPMENT COSTS			
	ESTIMATED COSTS, $M		
	YEAR OF ESTIMATE		
DEVELOPMENT AREA	1980	1982	1986
FUEL AND ASSOCIATED PROCESS DEVELOPMENT	45	37	19.3
FISSION PRODUCT BEHAVIOR	10	11	16.4
GRAPHITE DEVELOPMENT	20	35	10.9
MATERIALS (METALS, CERAMICS)	15		10.1
TOTAL	90	83	57.7

LW013

STATEMENT SUBMITTED FOR THE RECORD
SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION
COMMITTEE ON SCIENCE AND TECHNOLOGY
UNITED STATES HOUSE OF REPRESENTATIVES

RECENT PROGRESS AND FUTURE PLANS
HIGH TEMPERATURE GAS-COOLED REACTOR PROGRAM

JAMES M. KENDALL
DIRECTOR OF PROGRAM DEVELOPMENT
GAS-COOLED REACTOR ASSOCIATES

Madam Chairman and Members of the Subcommittee, my name is James M. Kendall, and I am Director of Program Development of Gas-Cooled Reactor Associates (GCRA), which is a utility/user organization that supports the development and commercialization of the High Temperature Gas-Cooled Reactor (HTGR). In referring to GCRA in my discussions, I am referring both to the GCRA staff and the member utilities as represented on the various committees and working groups that advise and direct the staff.

Prior to my current position, I served as a member of the GCRA Management Committee representing Yankee Atomic Electric Company. I also served as chairman of the GCRA Project Strategy Committee, working with other utility representatives during 1985 on what has evolved into GCRA's proposed project initiative. I am pleased to be with you today to discuss recent progress and future plans related to the continued development of High Temperature Gas-Cooled Reactor (HTGR) Technology.

GCRA has an active role in the DOE HTGR Program through numerous program integration and support functions as well as the development of utility/user requirements to guide the design effort. The fact that the that the design is meeting our demanding requirements is a primary factor in the growing utility support for gas-cooled reactor technology. Since the DOE HTGR Program activities are being addressed by the other members of the panel, I will focus on the activities of GCRA and its member utilities with regard to the utility initiative in support of a Demonstration Project, which is intended to complement and build upon the results of the DOE Program. This initiative was discussed briefly in the Joint Participants Statement on the HTGR Program to Congressional Oversight Committees submitted last February. A more detailed definition of the initiative entitled the Modular Gas-Cooled Reactor Project Strategy Plan was submitted to Secretary Herrington in July with copies to members of this Subcommittee.

The Project Strategy Plan was developed by GCRA over the past 18 months at a total cost approaching two million dollars. The purpose was to establish a draft framework for

Modular Gas-Cooled Reactor development which could serve as a starting point for developing a Final Plan acceptable to all parties. Although the Department of Energy and the program contractors have been kept informed and have participated in the evolving strategy over the past year, the draft Plan should be viewed as solely representing GCRA.

The primary foundation of the Plan is the information developed over the past year in the Project Definition Study. The Study, funded by GCRA with cost sharing support from the Program contractors, was initiated to establish an overall scope, schedule, cost estimate, and preliminary licensing plan for a Modular Gas-Cooled Reactor Demonstration Project. The work has been completed, and the final report is scheduled to be issued next month. Since the results of the Study are summarized in the Project Strategy Plan, I will provide only a very brief overview.

The reference Demonstration Plant site chosen for future work is the Test Reactor Area of the Idaho National Engineering Laboratory, with the intent of the operating entity, currently identified as the Utility Project Company, leasing the site from the Department of Energy. The plant would be licensed by the Nuclear Regulatory Commission, following a relatively conventional two step process. A full range of testing was identified for the first two years of plant operation, with particular attention to a series of response to accident tests. In addition to demonstrating construction and normal operation of a single module plant, the Project will be directed toward a demonstration of the passive response characteristics with the primary objective of instilling confidence in the safety and investment protection of the plant among utilities, regulators, the financial community, and the general public.

The cost of all activities required to complete and operate the Demonstration Plant was developed through a detailed code of accounts with preliminary results summarized in the Project Strategy Plan. The field direct cost estimate for the Demonstration Plant of approximately $1700/kw is very encouraging for a first-of-a-kind, single module plant. On the other hand, the indirect cost estimate for engineering is about $2000/kw. We are looking hard at how this can be reduced, but in part I believe it represents the high cost of the design and licensing process for a nuclear plant in the current environment. If the certification process is successful, the design and licensing scope and costs will be greatly reduced for commercial plants, so we feel these costs can be held to a competitive level for later plants. A schedule was developed under the assumption that the Project would not be constrained by the availability of funds. This schedule projects the start of full power operation in 1993. The schedule in the Project Strategy Plan was adjusted to reflect a more realistic, but still aggressive projection of available funding, resulting in full-power operation in 1995.

The Project Strategy Plan proposes a central role for the entity representing the electric utilities, acting as an involved utility owner in the design and construction of the Demonstration Plant. Consistent with this role, the utilities would provide a major portion of Project funding, and share Project risk with the contractors, with the federal government, providing supporting funds not to exceed 50% of total expenditures.

We recognize that establishing utility participation in a Demonstration Plant at the level envisioned presents a formidable challenge that has been greatly complicated in the near-term by the events at Chernobyl. I cannot say to you today that the utility support identified in the Plan will, in fact, be forthcoming. Much work remains to be done. However, the recent approval by the GCRA Board of Directors of a roughly 50% increase in member dues and a target 1987 GCRA budget consistent with the Plan, in conjunction with approving the submission of the Plan, can be taken as an expression of resolve by GCRA to move the Project initiative forward.

With regard to future plans, our most pressing objective is the evolution of the draft GCRA Project Strategy Plan into a Plan that has the full support of all elements of the HTGR Program. Continuing the technical development of the Project in 1987, GCRA plans to support the development of a conceptual design of the Demonstration Plant, initiate discussions with the Nuclear Regulatory Commission on plant licensing, and continue with Demonstration Project planning activities with emphasis on establishing the utility consortium to serve as the owner and operator of the Demonstration Plant.

As we understand the general planning within the Department of Energy with regard to civilian advanced reactors, narrowing to a single concept may take place during the next year or two. We also understand that a Demonstration Plant Project is being considered for the selected concept with a major role to be played by the industry and a supporting role by DOE. If industry, and in particular, the utilities, are to play a major role, it is obvious that the concept selection criteria must be focused upon effectively meeting utility needs for future power plants. Also, major utility support for a demonstration plant is not likely unless the prospective utility supporters play a major role in the selection of the concept.

One of the key issues associated with the Project Strategy Plan has been the credibility of the proposed industry support, and in particular utility support envisioned. We fully appreciate the difficulty of raising the proposed resources, and recognize that the proposed utility role is one of the most challenging elements. But the challenge is certainly not due to a lack of resources, but rather a question of the priority given the Project both by utility industry management and the regulating agencies. A sense of priority will not appear overnight, but must be developed over time by an increasing

sense of involvement in and ownership of the process. Regardless of the advanced nuclear technology to be developed, if any, the active, involved support of the ultimate users during these early days must be supported and encouraged if there is ever to be any hope of the industry led project the DOE envisions for the future.

Madam Chairman, the continuing strong support of this Subcommittee, particularly through the recent difficult times, has been an essential element of our success to date. We will be working hard to build upon that support to move the utility elements of our Plan forward. We hope to be able to report back to you next spring that we are all pulling together and succeeding in meeting the difficult challenges that face us.

Mrs. LLOYD. Thank you very much, Dr. Kendall.

I want to ask all of you one question. You can all reply starting with Mr. Northup.

What do you feel is the role of this committee for the future progress of the MHTGR?

What are we doing that is right; or what should we be doing that we aren't doing?

Mr. NORTHUP. I think starting, as I mentioned, in November 1983, when this committee said that things aren't right in the way that we were heading, ourselves, caused us to relook at ourselves and we were heading, and we came up with what we think is a terrific approach to getting nuclear back on path, both here and as Linden, in the underdeveloped countries. This is a tremendously exportable product.

I think that start can be looked back on as time goes on as even more profound if we are certainly successful. I think what this committee certainly can do is keep working with us, keep pushing us, that is both is as vendors, utilities, and the DOE, in resolve to bring the thing to fruition.

It is very hard to get the inertia going. It seems like, if I look over the last 10 years where we were on one program or another for about 18 months duration, and then suddenly we are yanked off in another direction, you just never get anywhere that way.

We have been on this direction now for 3 years. In the last year in a very, very focused way on one particular design, that seems to be meeting all the criteria that was set down.

I think we need to keep the train on the track, and pull the throttle out a little bit.

Mrs. LLOYD. Thank you.

Mr. Chernock.

Mr. CHERNOCK. Madam Chairman, I have been fairly outspoken in 1985 with respect to progress on this particular program. I believe my views at that time was let's quit generating paper studies, and let's get on with the job of getting with a demonstration project.

I believe that progress during the last year has been extremely good. There has been a focusing from 14 concepts down to 4, down to 1.

We now have a design concept that all participants agree on, are focused on and are prepared to move on.

Again, that is paper. If we do not have a demonstration project commitment, I don't see how we are going to achieve commercial viability of this important concept.

So, I would urge you to push very hard for a demonstration project for the Modular HTGR Program.

Mrs. LLOYD. Well, I share your feelings. But I don't want us to lose the momentum on this project, because if you do you are going to see the support in the Congress is going to dissipate and don't want that to happen.

First of all, we need to move on as rapidly as possible.

Paul.

Mr. KASTEN. I feel the basic objective of the subcommittee and committee is to give guidance and support to a responsible nuclear program in the United States, and that means that one has to look

at the longer term than just the near future. Of course, I know that you share the view that liquid power is necessary, and we do, too, that it is an important energy input to our energy needs.

Further than that, the modular HGTR seems to fill a particular niche at this given time. And proceeding with the demonstration project would be very important with regard to being able to get such a technology on the move. Keeping in mind, that in the long term, that the HTGR has more applications than just electricity production.

Being modular it can be sited in close to industry, and process heat could be utilized. In the long term it could be used for water splitting, if we ever go to a hydrogen economy.

So it has a long-term role, but you have to get started. This is a system that looks like it can be applied at an early date.

Mrs. LLOYD. I agree with your comments. I think that is another good reason why we need to continue to have oversight hearings on a more frequent basis.

Dr. Kendall.

Mr. KENDALL. I have to say I am relative newcomer to the overall process of the Government. My background has primarily been in the utility on one side of it or another.

From what I understand of where the program has been, I think that the role of this subcommittee is to continue to do what you have done so well in the last few years of being somewhat an impatient oversight committee looking at what the process is, the progress, and continuing to challenge both Government and private-sector participants to move forward. From our standpoint we realize that we have a difficult road to try to bring up the level of participation on the utility side.

I think the activities of this subcommittee is a real help. So I see continuing what you are doing as being a vital part of bringing this whole thing off.

Mrs. LLOYD. Thank you.

Mr. Morrison.

Mr. MORRISON. Thank you.

I would like to discuss briefly with you gentlemen on this panel, about the fuel process. Is there any reason to be concerned at all with the integrity of this fuel kernel, it seems to me that this system is only as good as that coating is precise; am I in error in that observation?

Mr. NORTHUP. I believe you were out when I mentioned poppy seeds; were you not? Were you here when I was talking about them?

Mr. MORRISON. I came in just as you concluded.

Mr. NORTHUP. Let me try to put the picture in your mind as I see it. If you envision a very large secondary containment structure that is around a conventional nuclear plant, and shrink that down around a particle of fuel that is about the size of a poppy seed, and then you take three more structures like that and shrink those down around this fuel particle, if you then imagine that that shrunken structure can handle pressures that are 50 times the capacity of a containment structure, I think you could get a grasp of how significant the MHTGR fuel is. And then we build the rest of the system so that containment structure is never violated.

These thicknesses, the strength, the density, the process, everything about these structures is a design feed, it is not just that you take a piece of uranium and paint it with a brush. They are formed at over 2,300 degrees centigrade, and the worst accident condition we have in the whole reactor, even with loss of coolant, seismic, whatever, may take us up to 1,600 degrees centigrade. There is an enormous factor of safety about the level that they will experience to what they were actually born with.

The experience has been that we have produced millions, in fact, we have operated in reactors billions of these structures. We have examined their life history to full burnup.

We have measured their capacity for this retention feat, and we have a large amount of data that shows that they do the job intended. Fort St. Vrain, THGR in Germany, AVR reactor that has run for 18 years with this type of fuel, in fact, much worse in the early days when the structural design was not as sophisticated as we have become in being able to do it, even there the amount of release from these particles has been very, very small.

The reason that the Oak Ridge National Laboratory does not have several hundred million dollars of research work in this area is that what we need now is simply to make sure that we have this type of coating fully defined at the high level of quality that we want it for the modular reactors.

I am very convinced that the particle will do its job. We have been seeing it do its job for the last 20 years.

Mr. MORRISON. OK.

Mr. KASTEN. May I comment?

Mr. MORRISON. Please do, Paul.

Mr. KASTEN. I would like to add a perspective on that. Back in the mid-1950's, and the late 1950's when these coated particle fuels were being developed, of course, they didn't start out as good fuels. And my perspective was that this is rather impossible. We are going to end up with billions of particles and put coatings on them and they are going to flake off and not going to hold up.

But there has been a remarkable development. Through the years those coatings have gotten better and better. Information has been obtained in terms of how you get good performance.

The key is put down a high-quality coating, which in the case of power carbon means it is isotropic, it has uniform properties. In the case of silicon carbide it has to have a high density and fairly uniform properties.

But this has been developed through the years. Another key is that you start out with spherical particles and you keep those particles spherical during the fabrication process so the quality control during a fabrication is important.

But it has been demonstrated that this can be done as Gene Northup mentioned, on millions, billions of particles. It has been done here and in Germany. It is a very successful operation and we know how to make good coated particle fuels that retain fission products.

Mr. KENDALL. Could I make one comment on that?

Mr. MORRISON. Sure.

Mr. KENDALL. I would say from the utilities standpoint, the requirement of no emergency planning, no sheltering, which is a part

of the utility requirements, places a very stringent quality requirement on the fuel. I think that there is certainly indication that that can be met, but I think it is going to be very challenging to develop the bulk fuel manufacturing capability to maintain those requirements.

From the utility perspective, as I see it, in going around and talking to utilities, the prospect of signing on knowingly, basically utilities did not sign on knowingly to emergency plans and the tremendous political problems that that causes. I think the question of emergency planning requirements is a vital one.

So, we have to have that. I think that while we have seen very good performance with the fuel, we have to remember we placed a very tight spec, and it is fundamental to meeting those requirements.

Mr. MORRISON. Well, I asked the question particularly because I looked at a lot of different fuel elements and know the precise requirements, and when you are dealing with millions of small kernels, I wanted to ask the question. You have been most reassuring.

Do you anticipate any different treatment approaches as far as waste from these fuel elements?

Mr. NORTHUP. I am glad you asked because I wanted to add that to my particle—you get me warmed up on this. Since it is in such a vessel and it is ceramic or glass like, the storage of the fuel becomes captive to its own house. Therefore, we do not have to run it through another process, reduction process.

It is a once-through fuel that we have for the low enriched, we are in the low enrichment or below the low proliferation limit. We would simply store these, and they could be stored in the fuel block themselves, or what we think is even better, we drop the fuel rods out of the large graphite blocks, the large graphite blocks then can go to very low waste disposal.

Whereas the particles in their formed rods then can be stored right within their own shells; we do not have to reincapsulate them, or reformulate them, or run through a chemical process.

Mr. MORRISON. But they could be?

Mr. NORTHUP. We could be. The fuel was specifically designed in the early days for reprocessing, the separation of thorium particle, as a separate particle; the uranium particle is a separate particle, is for that reason, of allowing us to do recovery later.

Since we have dropped to low enriched, and we are not producing plutonium in these cores of any appreciable amount, there is no economic incentive to do that. If we did go back to a high enriched economy then reprocessing becomes more attractive in the economical sense, then we can certainly do that.

Mr. MORRISON. Just one last question.

Dr. Kendall, do I understand from your testimony that the site at Idaho has been pretty well selected by the associates for a reference for a demonstration plant; that will be your recommendation?

Mr. KENDALL. That is correct. The basic demonstration plant study looked at two different kinds of sites, one being repowering of an existing utility site. The other being of a more remote site starting from a green-grass situation.

In looking at all the various factors, we ended up with the Idaho site as our reference for moving forward. The model itself would be

a utility entity that acts as the owner-operator of that plant on a leased site through DOE. Yes, we intend to move forward in our planning at this point with the Idaho site.

Mr. MORRISON. I said that was my last question, but I would like to ask one more.

I would like to have a reaction maybe from any of you that want to react. The Department of Energy has actually requested only a certain amount of money for 1987, which is the budget we are wrestling with now.

Do you interpret that as a lack of enthusiasm on their part, or do you feel let down, from your point of view in private enterprise?

Mr. NORTHUP. We need, I think, desperately to be supported for the $25 million for fiscal year 1987, that has been laid out by the Senate. Otherwise, coming off a year where that is about what we are applying in 1986, it is going to be a drop, and a sizable drop if we go to the lower number.

Certainly, with DOE coming out as a lower number, $5 million, there is certainly no way that the momentum can be carried with that type of funding in 1987. We understand the difficulties in DOE and their budget formation.

We understand the help that we get from this committee in getting those numbers jacked up, but you hate to go through that process year after year, and we certainly would like to see a better chunk of the money applied to this program. But we have to be disappointed, discouraged—we have had hills to climb and doors to open before, and since we have got such a good product we are not going to get so discouraged that we quit, but it certainly would keep the momentum going at the levels we have established.

Right now, at this stage, although Warren is talking about producing paper, this is a paper that will carry this thing to fruition. It is not paper that we are putting on the walls or something to the trash, it is setting the design, the design drawings, these come out of the minds of men that we have had working in this field for a long time.

There is where the product is produced, and then execute in the field and the shop for fabrication.

Mr. CHERNOCK. Can I comment on that?

As far as paper in the past, what we have had prior to 1985, has been, as I expressed in testimony at that time, a program which was moving in a variety of directions. I firmly believe that well-constructed paper trails leading to the current design selection is extremely important.

So my comment on paper is let's not have paper for 14 different concepts. Let's have paper for one concept; know how we got there; why we got there; and where we are; and now let's take that concept and let's go ahead and demonstrate it in an actual operating demonstration plan.

In order to do that I do not think that we can afford to come back each year and start beating on people to increase a budget request. I think it is important to recognize now that a demonstration plant ultimately is going to be required, and to work in that direction with respect to the budgetary process.

Mr. KASTEN. One comment in terms of base technology development. The $5 million budget certainly would disseminate any base

technology work. In particular, an important item there is demonstrating fuel performance in simulated fuel elements.

We spoke very highly of the performance of coated particles. We still do need to demonstrate—there is information that says that it can be done.

There is some preliminary information that is encouraging, but we really need to proceed on demonstrating the fuel element, in addition to the coated particle. So, we feel that the $25 million budget would be really necessary to move forward in the program to carry out both the conceptual design as planned and the base technology program in a meaningful way.

Mr. MORRISON. Great.

Mr. KENDALL. If I could just speak briefly to that. It has never been clear to me what kind of a scope goes with $5 million, and so I think obviously it would be very difficult to keep the program going.

It makes it difficult for us from two standpoints, one being in terms of talking to the utility, they want to know why is this, why do we have this level. The other being that we have spent a lot of energy trying to get it up.

I realize we are short on time; I would like to say more.

Mr. MORRISON. Thank you all very much.

Thank you, Madam Chairman.

Mrs. LLOYD. Thank you.

Mr. Stallings.

Mr. STALLINGS. Thank you, Madam Chairman, I do not have any questions.

I was going to ask the same one that my colleague from Washington asked, about the Idaho siting. I have to agree with Dr. Kendall's wisdom in selecting that site. I think that shows how thoroughly you are doing your homework.

Mr. MORRISON. I didn't say we agreed with it.

Mr. STALLINGS. When you review their documents, you will.

Mrs. LLOYD. I want to thank all of you for your part in this hearing. I don't think we have ever had a better hearing. I think it has been good. I want to commend the staff, Ron Williams and all of our staff members that are here, for their part.

So often when we bang the gavel down we don't thank your staff. But we do have excellent staff on this committee.

The witnesses and the Members, that did participate today, I failed to mention earlier, I would like to mention now that we will be sending questions to all of the participants we would ask you to respond as quickly as possible. Because certainly we can't cover all the questions we wanted to ask today. But I think it is pertinent; it is important to keep this program on track; and to the best of our ability, hold hearings again next spring to move on with the MHTGR Program.

If there is no further comments, thank you again.

The subcommittee is adjourned.

[Whereupon, at 12 p.m., the hearing adjourned.]